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OAK RIDGE, TENNESSEE 37830

December 17, 1980

Dr. Michael Tokar
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mike:

Subject: ORNL Calculations of Postulated Fort St. Vrain Reactor LOFC/FWCD Accident Response for Core Support Thermal Stress Evaluations

Background

The purpose of this letter is to describe the methods used in the subject analyses, summarize the results, and evaluate the accuracies of the calculations.

In May, 1978, audit calculations of several worst-case postulated Fort St. Vrain (FSV) loss-of-forced convection (LOFC) accidents following a design-basis earthquake were performed under ORNL's NRC/LSR-sponsored HTGR safety program.[1,2] Subsequently, Prof. Theophanous of Purdue noted that during the firewater cooldown (FWCD) phase of the accident, the predicted temperature differences between certain lower reflector and core support block (CSB) nodes for adjacent refueling regions were very large (up to ∿ 1500°F). There was thus some concern that such high thermal gradients could cause large thermal stresses in the support block regions. The task of calculating these stresses, given the output of the ORNL ORECA code[3] calculations, was assigned to LASL, and their results and conclusions are reported separately.[4] Public Service Co. of Colorado (PSC) has also submitted an assessment of the problem.[5]

The large temperature differences predicted at the bottom of the core result from the uneven region (axial) temperature profiles that are generated during the earlier (LOFC) portion of the accident. Those regions with a high region power peaking factor (RPF) experience large reverse (upward) flows which transport the heat towards the top of the core. Then after the cooldown begins, the forward flows drive this heat downward, temporarily raising the temperatures at the base of the core to a much greater degree than those low-RPF region that always had downflow. The uneven cooldown is accentuated by the fact that the low-RPF region FWCD flows are larger ($\sim$ 2:1 or 3:1) because of the lower effective flow resistance that results from their lower temperature (and hence lower gas viscosity and larger density). LASL also noted the possibility of high CSB stresses caused by sudden cooling from reverse flows that occur in some regions $\sim$2 hours into the FWCD.

Some of the earlier LASL calculations indicated the possibility of relatively high stresses in the CSB for the reference 105% power case with the EQSB3 core[2]. In order to evaluate the potential problems for present FSV operating conditions, a second set of initial operating parameters was generated by GA based on a maximum power of 72% and worst-case peaking factors and outlet temperature dispersions for a cycle 2 E.O.L. core. The analyses described include both of these cases.

Description of ORECA Code Features

ORECA simulates the 3-D thermal-flow transient behavior of the FSV core. The ORECA core node structure models the 37 refueling regions and an 18-channel approximation of the side reflector with 8 to 10 axial nodes each, for a total of 440 to 550 nodes. The model accounts for variable flow distributions between refueling regions (including reverse flow in individual regions), approximates the heat transfer and friction characteristic changes with flow regime, and includes expressions for gas and core material physical property changes with temperature. ORECA is alternatively run as a stand-alone code (as in the present case), deriving time varying input values of total core power, flow, pressure, and inlet temperature from other sources; or ORECA can be run as a part of an ORNL overall system code ORTAP.[6]

The results of ORECA calculations have been compared with data from 4 FSV scram tests[7] and in numerous cases with output from the GA RECA code[8], and the comparisons have generally been very good.

The detailed features of the ORECA code have evolved somewhat since the original calculations were made. First, there have been numerous improvements and corrections made to the code through "normal use" both in-house and by others, lesson learned from verification activities, and a comprehensive review by BNL. Second, the original intent of ORECA was more to predict outlet gas temperatures and maximum core temperatures rather than core support block temperatures. Hence a more detailed node structure was implemented. Where originally one axial node was used for each region's lower reflector and core support block, the present version uses two nodes for the lower reflector and a third for the CSB. The new version of ORECA also calculates and prints out the heat flows into selected nodes via conduction and the heat flow out from convection. This information is used as input to the stress analysis code (by LASL).

Regarding the difficult question of interest in licensing matters, especially, as to how accurate the code's predictions are for postulated accidents, the following approaches have been taken:

1. The code's models were developed from generally well-understood first principles to avoid problems with misuse of empirical models outside of their expected ranges. The problems of modeling distributed-parameter systems with lumped-parameter approximations were also addressed specifically[9].

2. Wherever possible, internal consistency checks are made.

3. Peer reviews were conducted both within ORNL and by others.

4. The code has been exercised for over 5 years on a variety of transients, large and small, and the results scrutinized by many people. A key means of checking and understanding the code behavior and accuracy is through the use of sensitivity studies, which are especially easy to run on ORECA because of its relatively simple input structure and low running times and costs (∿ $5/run typical). This allows the investigator to alter the model and parameter assumptions and see directly how they affect the results.

5. The results of ORECA have been compared on benchmark-type problems with other similar codes, including RECA and FLODIS[10] and the agreement is generally good. The differences that do exist can usually be rationalized based on the known differences in the codes.

6. ORECA output has been compared with FSV transient data, primarily 4 scram tests[7] during cycle 1, and the agreement has generally been very good. Our program is continuing in its efforts to verify ORECA (and other codes) as much as possible with existing data and via proposed special tests.

ORECA Code Calculations of the Postulated LOFC/FWCD Accident

The original FSAR scenarios for the postulated design-basis earthquake LOFC accident stipulated that the last-resort firewater cooling system for driving the pelton wheel turbines on the main circulators would be operational without any delay. NRC subsequently determined that up to a 90 min delay should be allowed in the accident reanalysis. The natural circulation flows that would occur in the core within this period are sufficient to redistribute the heat significantly. Thus when forced circulation is restored, the distribution of the region cooling flows tends to aggravate the core temperature nonuniformities, since the higher flow resistance of the hotter regions (which need more cooling) restricts their flow more. Predictions of the maximum temperature differences in the lower core support regions, which occur ~2 hr into the FWCD, turn out to be quite sensitive to factors which affect the relative cooldown rates of the hotter and cooler regions, such as assumed total refueling region flow and the configuration of the region orifice positions. The total flow assumption depends both on the estimate of the FWCD system output and the fraction of the total flow that bypasses the core. A low-resistance core orifice configuration assumption adversly affects the flow redistribution, because the wider-open the orifices, the less effect they have on the flow distribution relative to the region temperature conditions. Hence in the reference case ORECA runs, the most conservative condition was assumed, i.e. the widest-open orifice was fully open.

The ORECA code predictions of maximum support region ΔT's have changed significantly since the original calculations for several reasons:

1. Using the more detailed nodal structure, the CSB is now a separate node which has a ratio of heat transfer area to heat capacity that is much smaller than the average for the lower reflector-CSB region; hence it doesn't cool down as fast in the FWCD phase of the accident as the node which originally represented both the lower reflector and the CSB, and the maximum ΔT's between CSB's are much smaller.

2. The earlier version of the ORECA code had no printout which indicated the region orifice positions that would be required to give the individual region flows for a given overall core pressure drop. At one point in this analysis, some of the results were generated using unrealistically small core pressure drops, i.e. the widest-open orifices would have had to be open more than 100%. When more realistic values were used, the predicted maximum ΔT's were smaller.

3. Most recently, it was noted by LASL that the axial nodal region for which the maximum ΔT should be monitored is region 10 (the CSB), rather than region 9 (the bottom half of the lower reflector). Previously we had assumed that the ΔT's in the lower reflector, which were much larger, would have a more significant effect on the local stresses at the top of the CSB surfaces.

4. At the November 7, 1980 meeting at NRC it was noted that there were major differences between the CSB thermal conductivity expressions used by LASL and ORNL. LASL was using data for PGX graphite supplied recently by GA[11], while ORNL was using a relationship given for the conductivity of the ~~lower~~ reflector in GA-LTR-1[12]. Figure 1 shows a comparison of the radial conductivity functions used in the two cases. Since the LTR-1 curves are for heavily irradiated graphite, and the CSB's don't receive much radiation, it was assumed that the PGX data was more appropriate. This means the assumed CSB conductivity is $\sim$ 2-10 times greater in the newer version, and since interregion conductivity is a relatively important factor in the CSB heat transfer during the low-flow periods, the change resulted in much smaller values of maximum ΔT.

Some results of the reference runs for both the case of the 105% power EQSB3 core and the 72% power cycle 2 EOL core are shown in Figs. 2-5. In the first case, the radial regions with the maximum CSB ΔT's are 19 and 36, and in the second, regions 35 and 36. In the two cases, the maximum ΔT's between adjacent CSB's (axial region 10) are 657 and 602°F. For the nodes representing the bottom halves of the lower reflector (axial region 9), however, the maximum ΔT's are much larger: 1417°F in the 105% power case, and 1239°F in the 72% power case.

Sensitivity Studies

A number of sensitivity studies have been done during the course of this analysis, and it has been found that the maximum predicted ΔT's between adjacent regions vary significantly when some parameters are varied within what may be considered error bands that represent ranges of uncertainty. This is particularly true of variations in ΔT's for the nodes representing the bottom halves of the lower reflectors, which, however, may not have a major effect on the calculated thermal stresses of the core support structure.

Those parameters tested which did not have significantly large effects on the results were changes in overall core specific heat and thermal conductivity, laminar flow gas-to-core heat transfer coefficient, friction factors, and afterheat.

Those parameters which did turn out to be the sensitive ones were the assumed FWCD flow that cools the refueling regions, CSB thermal conductivity (major increase), core flow resistance (i.e. orifice positions), and initial core power level. Reductions in the LOFC period also helped.

Because of the variety of mechanisms involved in core region transient behavior, it turns out that as a parameter such as the assumed region FWCD flow is varied over wide ranges, different region pairs experience the greatest ΔT. Furthermore, the predicted maximum ΔT for a given pair of regions peaks out at a given assumed value of FWCD flow, while the peaks for other pairs occur at different flows. For example, in the 105% power EQSB3 case, FWCD flow sensitivity studies show that a maximum CSB peak ΔT of 771°F would occur between regions 19 and 36 if the flow were 0.8 of its reference value (vs 657°F for the reference flow). On the other hand, the peak ΔT between regions 20 and 21 increases from 602°F at the reference FWCD flow to 651°F at 1.2 times reference flow (Fig. 6). The studies also showed that the maximum ΔT between lower reflector nodes was 1659°F and occured for 0.8 flow, (vs 1417°F at the reference flow), (Fig. 7).

In previous calculations where the CSB thermal conductivity expression was the same as that used for the lower reflector graphite, the maximum ΔT's for adjacent CSB's (105% power case) was ~1020°F, compared to 657°F for the case of the new reference expression for PGX graphite conductivity. An assumed reduction in the new CSB reference conductivity of 20% gave a maximum ΔT of 702°F, (7% increase).

A sensitivity run was also made to show the effect of initial core ΔP on maximum CSB ΔT. For a 20% higher ΔP (representing a relatively high resistance core, with maximum orifice openings of ~50%), the maximum ΔT between regions 19 and 36 CSB's for the 105% power case was 566°F, or a reduction of 14% from the reference case. However, another pair of neighbors (regions 20 and 21) had the maximum ΔT, 608°F, which was larger than that of the other pair but still 7% smaller than the reference case ΔT.

The possibility of and severity of high CSB stresses caused by sudden cooling from region flow reversals during FWCD would depend on the magnitude of the reverse flows and the differences in CSB and outlet plenum temperatures at the time of reversal. The reverse flow phenomenon is relatively difficult to predict accurately, however, since these flows are set up by small driving forces that depend on the axial temperature gradients in the core. The ORECA code predictions indicate that reverse flows during the FWCD period are much more prevalent in the 72% power case than in the 105% power case. In the former, the first predicted reversal occurs 2.5 hr after the start of the FWCD, and 4 hr after the start of FWCD there are 4 regions in reverse flow. The magnitude of the reverse flows are also quite sensitive to the assumed value of FWCD flow. In the 105% power case, the first reversal doesn't occur until 4.5 hr after the FWCD, and only that one region (region 20) ever reverses (at least in the first 6 hr).

Summary and Conclusions

Calculations of core temperature transients during the postulated design-basis earthquake LOFC/FWCD accident sequences were made for two sets of initial condition data: 105% power with an EQSB3 core, and 72% power with a cycle 2 E.O.L. core. Summary results were presented in this report, and detailed outputs from the ORECA code showing core block temperatures and pertinent heat flows were sent to LASL to be used as inputs to their stress analysis codes.

Sensitivity analyses were also run that showed how the predicted core support region temperatures were affected by variations in the assumed run conditions, models, and parameters.

Estimations of the accuracy of these predictions should consider the following limitation of the code:

1. No model verification tests have been done (and none are likely) which would confirm the redistribution of heat in the core, especially the extent of the shift to the top, during an extended LOFC. This first stage of the accident is crucial in that it sets up the conditions for the "race" between hotter and cooler region cooldowns during FWCD.

2. Even though the ORECA model uses many nodes to approximate the core temperature distribution, the lumping is gross considering the need to calculate thermal gradients in specific block regions. The ORECA models also do not account for interregion bypass flows, estimates of which are incorporated into the LASL stress calculations. The effect of thermal radiation heat transfer from each CSB to the lower plenum was also not included in the ORECA models, although its omission would probably make the calculated CSB temperature dispersion larger.

3. Probably the greatest uncertainties in the calculation are the starting times and the extent of the reverse flows developing during the FWCD phase. If these turn out to be crucial factors in determining potential CSB damage, code verification tests such as those proposed by ORNL[13] would be recommended.

4. On the other hand, there are a number of factors favorable to the credibility of the calculations that were enumerated previously in the section on code description. Considering these factors, I believe that the calculations made for this investigation closely approach the best estimates that could reasonably be made in the absence of much more extensive testing and code verification efforts. Under the circumstances, however, considering the uncertainties, there is no reasonable way to assign error bands and uncertainty estimates to the final results.

Please let me know if you have any further questions or comments.

Yours truly,

Syd Ball

S. J. Ball, Manager
HTGR Safety Studies for NRC/RSR

SJB:rtw

cc: C. A. Anderson-LASL
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J. C. Conklin
Ron Foulds-NRC
R. M. Harrington
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P. R. Kasten
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J. P. Sanders
P. M. Williams-NRC

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Fig. 1 Comparison of Radial Conductivity Function for lower reflector[12] and PGX graphite[11]

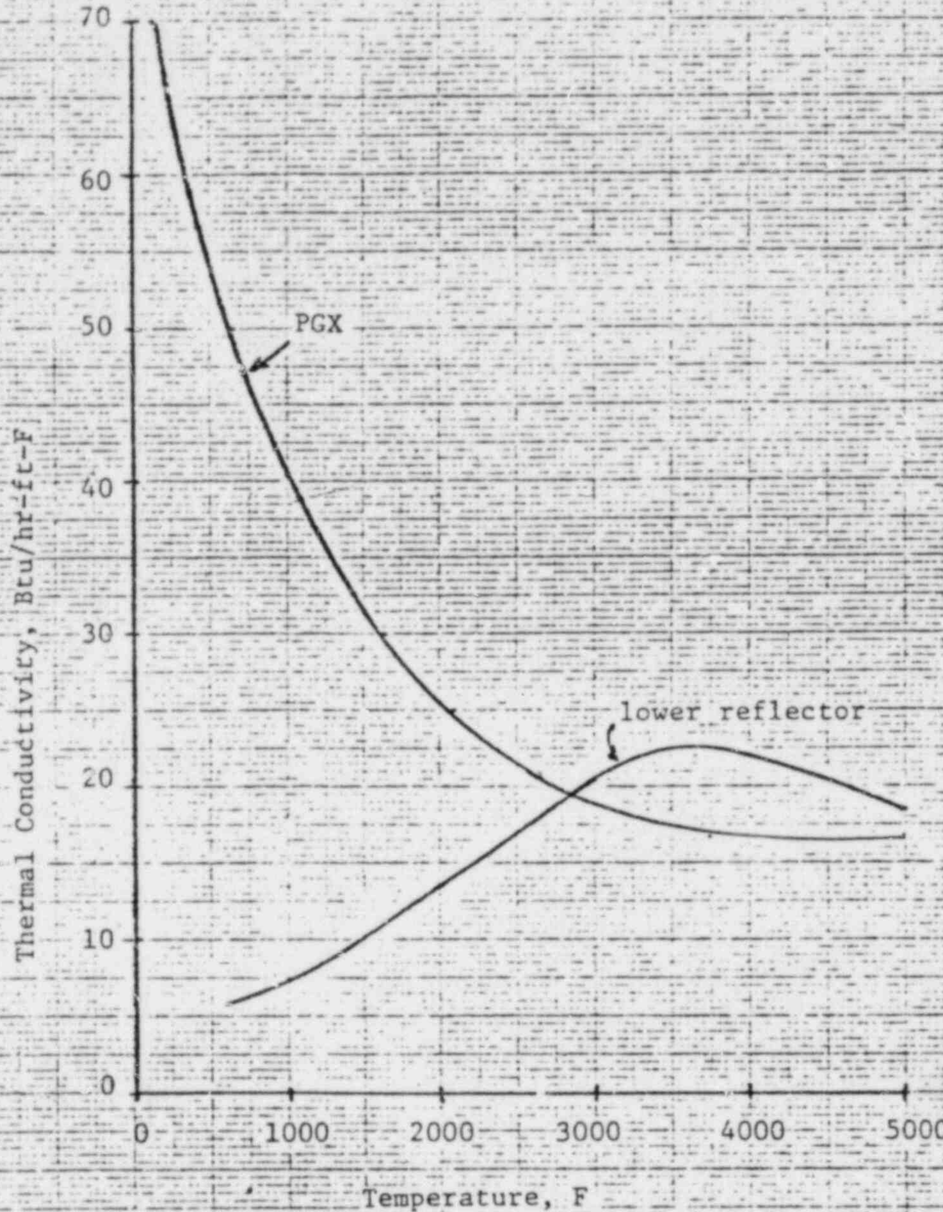


Fig. 2

FSV LOFC/FWCD Core block Temp 105% Power.

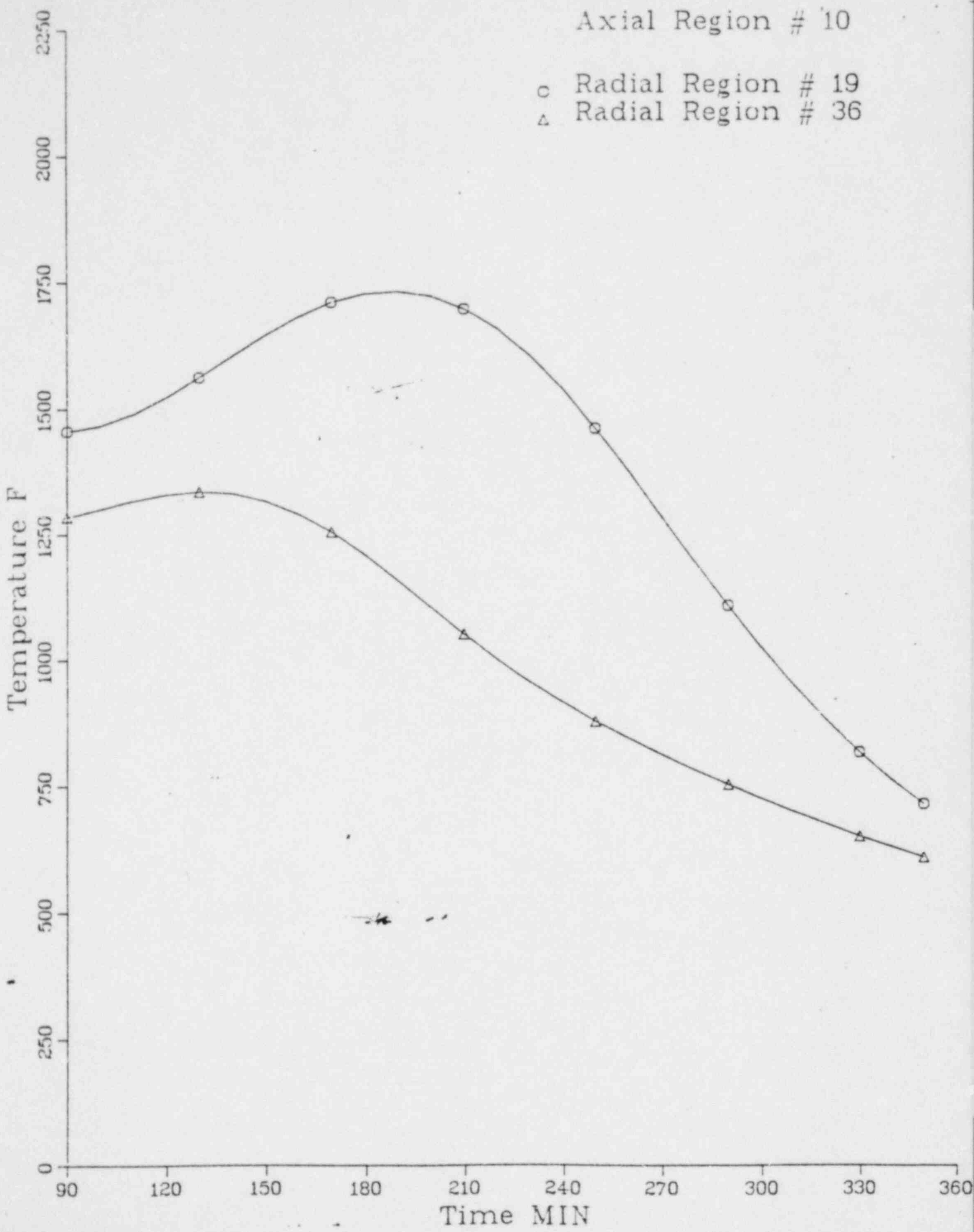


Fig. 3

FSV LOFC/FWCD Core block Temp 105% Power.

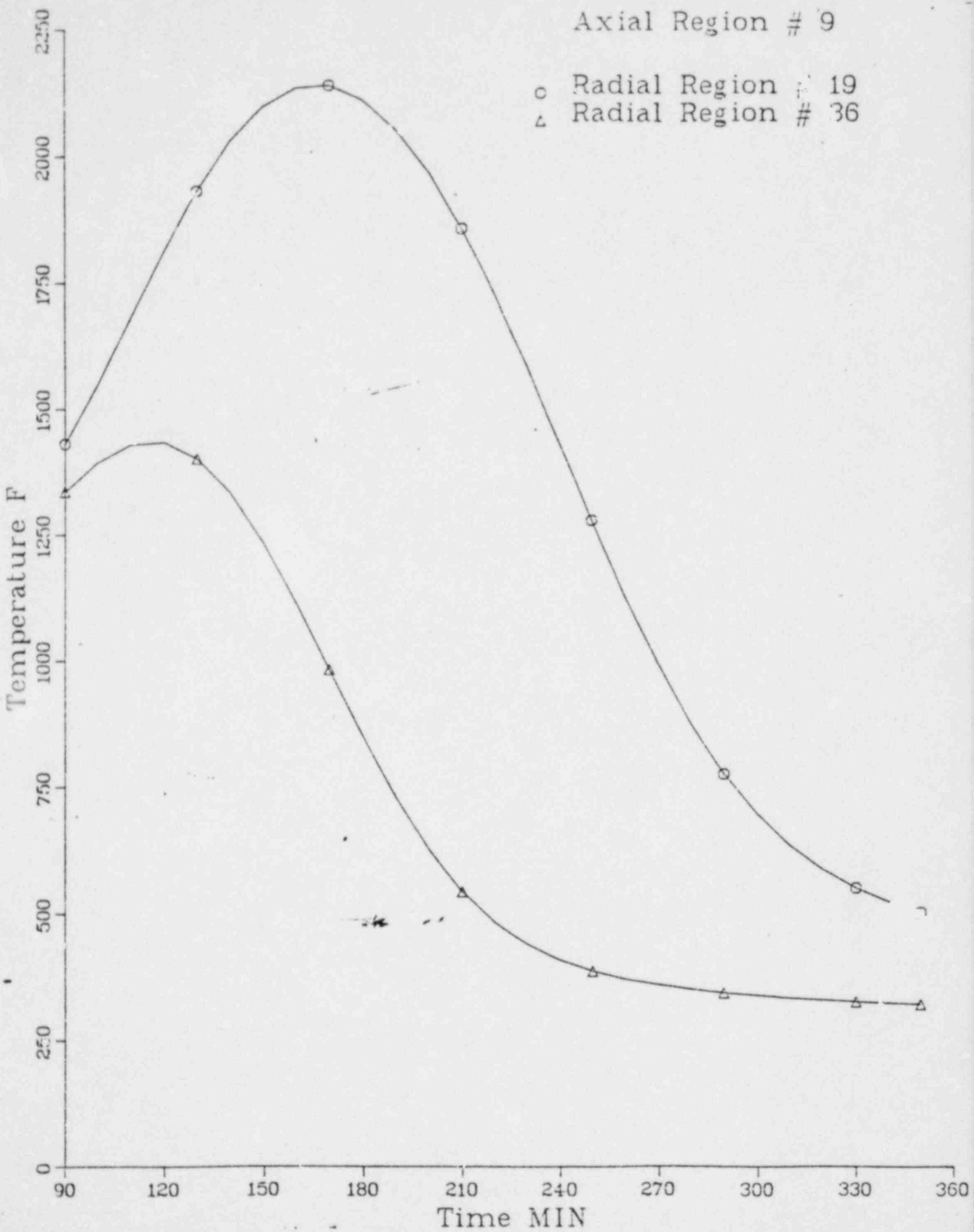
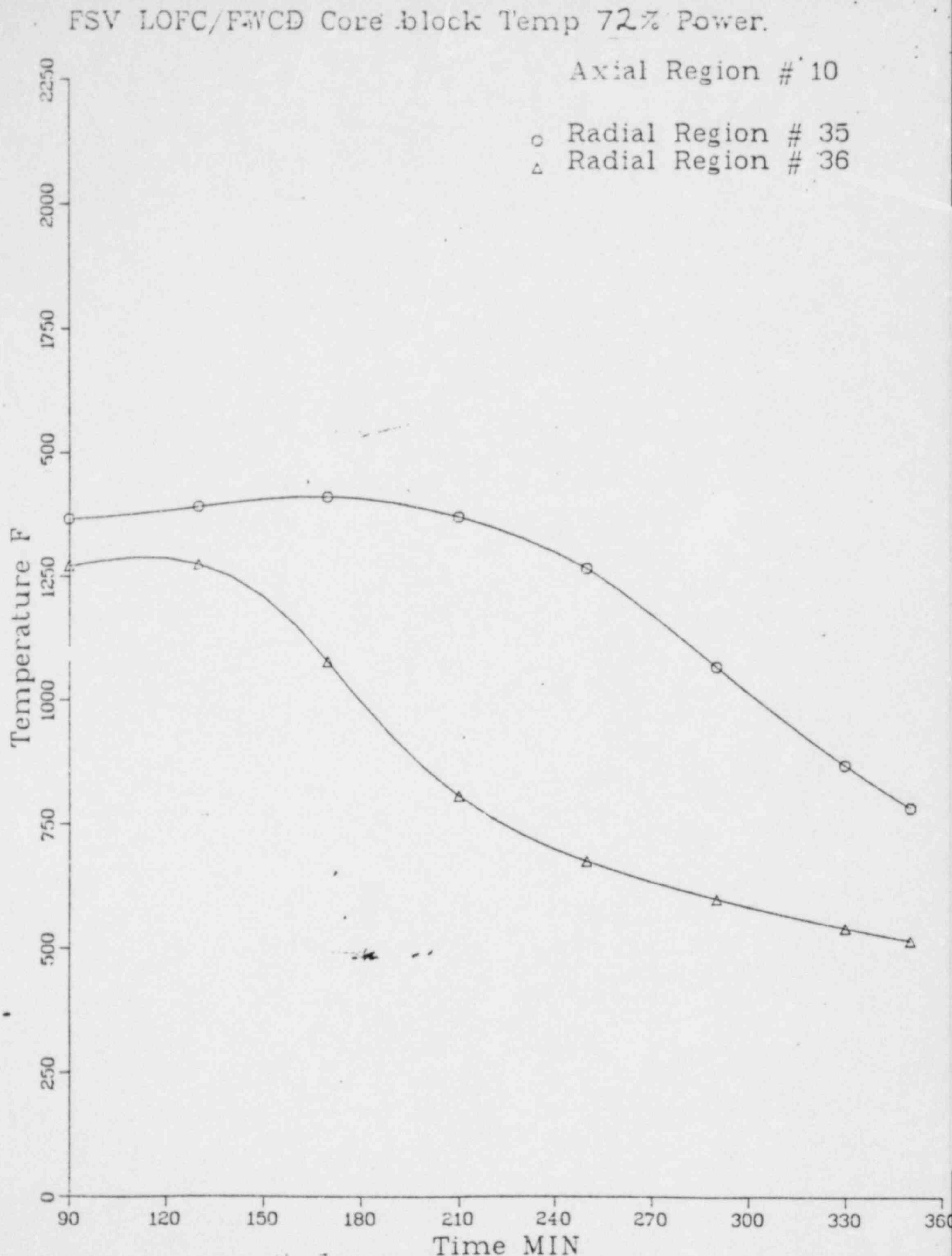


Fig. 4



FSV LOFC/FWCD Core block Temp 72% Power.
Axial Region # 10
Radial Region # 35
Radial Region # 36
Temperature F
Time MIN
0
250
500
750
1000
1250
1500
1750
2000
2250
90
120
150
180
210
240
270
300
330
360

Fig. 5

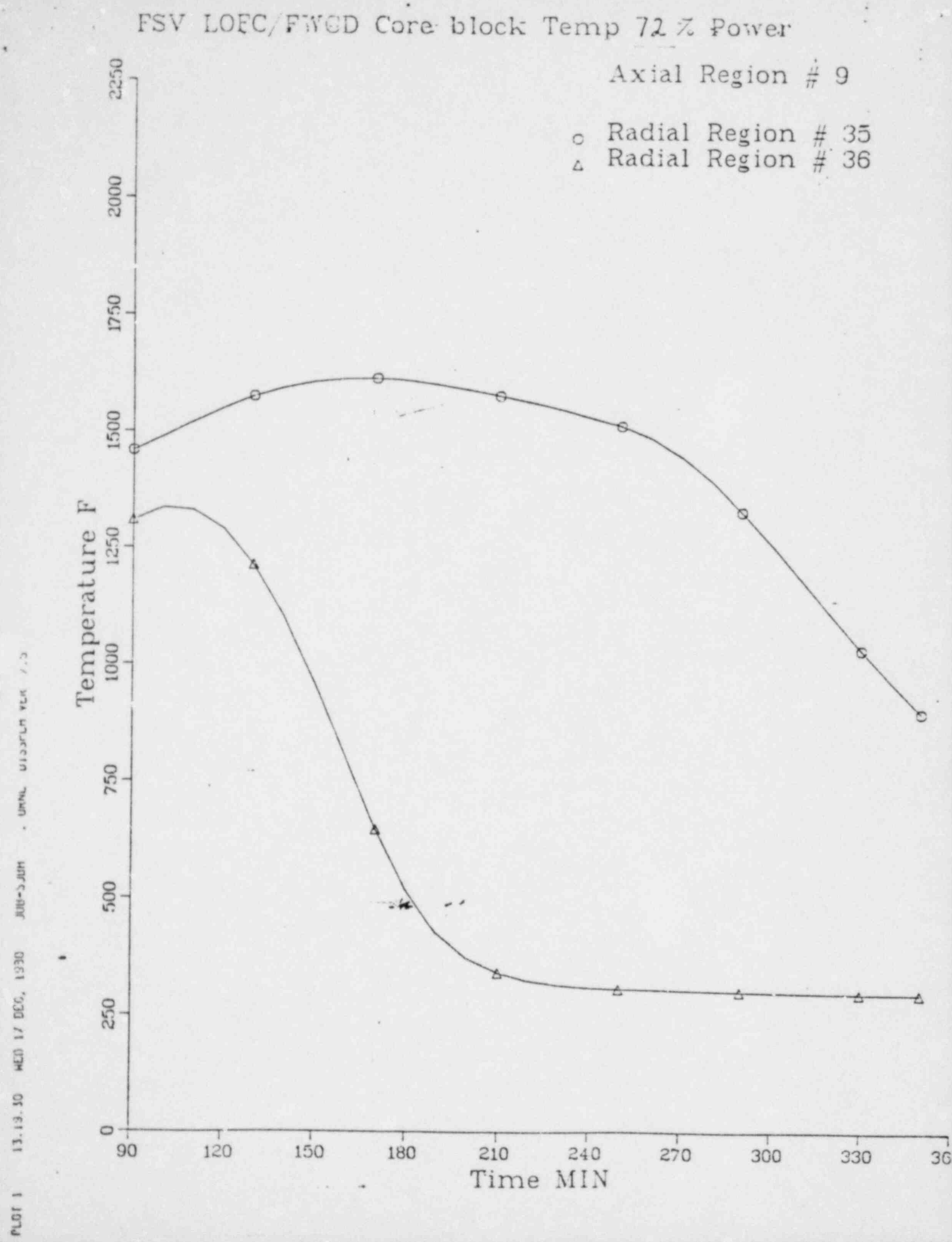

FSV LOFC/FWCD Core block Temp 72 % Power
Axial Region # 9
Radial Region # 35
Radial Region # 36
Temperature F
Time MIN
0
250
500
750
1000
1250
1500
1750
2000
2250
90
120
150
180
210
240
270
300
330
36
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Fig. 6

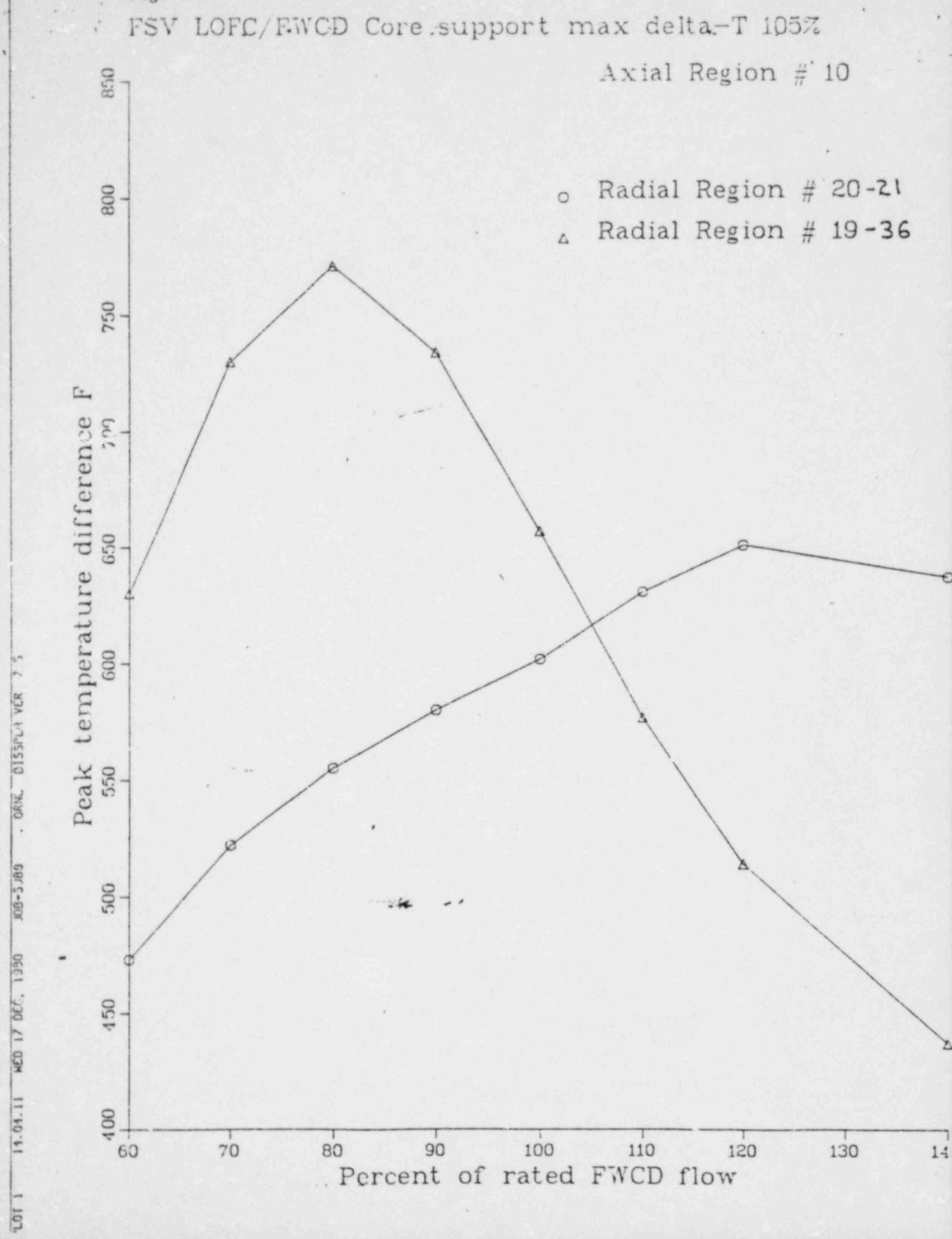
FSV LOFC/FWCD Core support max delta-T 105%
Axial Region # 10
Radial Region # 20-21
Radial Region # 19-36
Peak temperature difference F
Percent of rated FWCD flow
400
450
500
550
600
650
750
800
850
60
70
80
90
100
110
120
130
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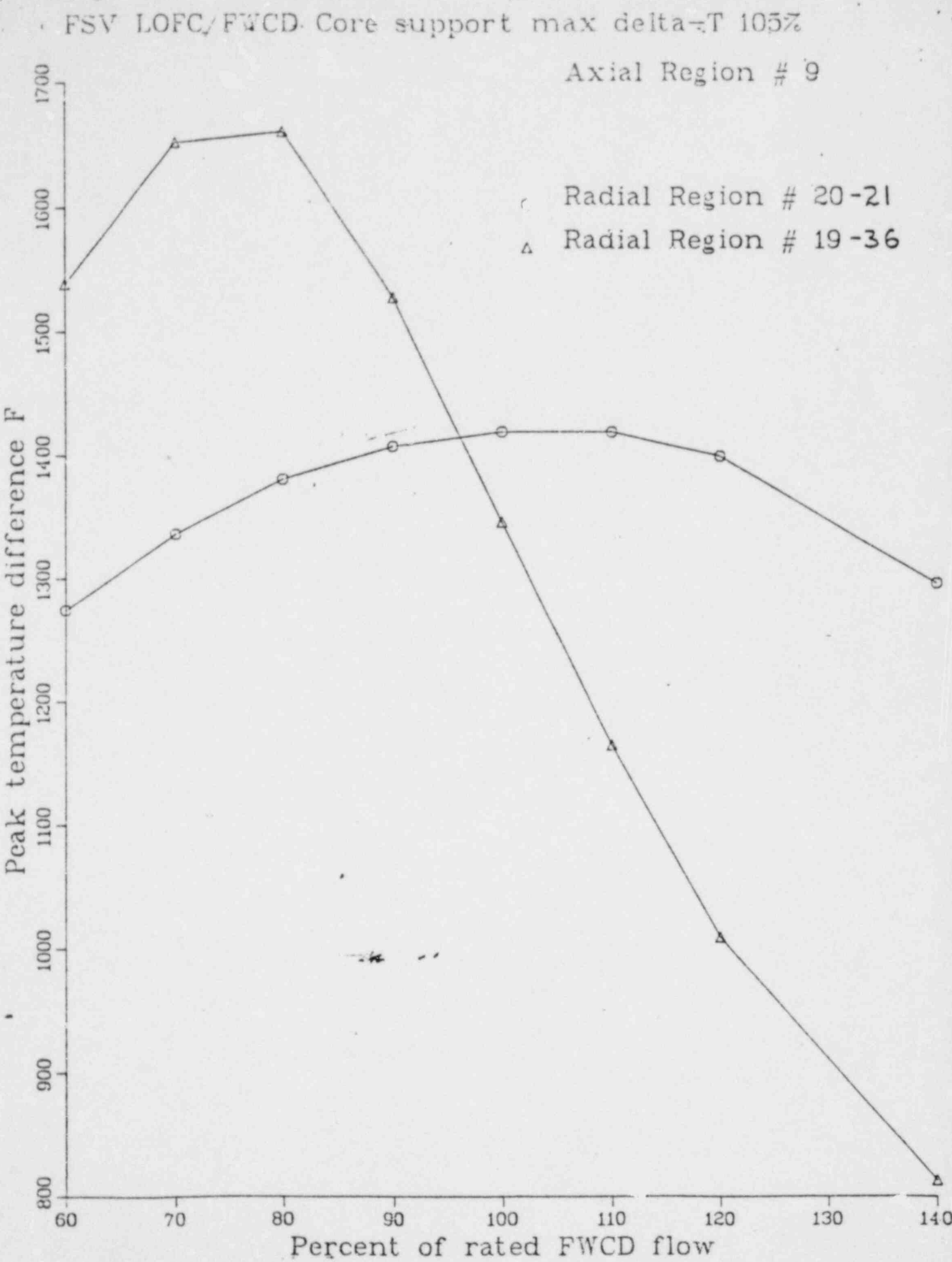
Fig. 7



FSV LOFC/FWCD Core support max delta-T 105%
Axial Region # 9
Radial Region # 20-21
Radial Region # 19-36
Peak temperature difference F
Percent of rated FWCD flow
1700
1600
1500
1400
1300
1200
1100
1000
900
800
60
70
80
90
100
110
120
130
140

OAK RIDGE NATIONAL LABORATORY
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December 3, 1980

Dr. Charles A. Anderson
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Los Alamos, New Mexico 87544

Dear Chuck:

Subject: ORECA Calculations for FSV Thermal Stress Analyses Using New GA Thermal Conductivity Data for Core Support Block

Enclosed are the latest ORECA code calculations of the postulated 90-minute LOFC-FWCD accidents, one for the 72% power cycle 2 E.O.L. core and another for the 105% power ECSB3 core. The difference between these and previously submitted cases is the use of the new relationship for core support block (CSB) thermal conductivity obtained from reference 1 (enclosed). Other core component conductivity relationships were taken from Fig. 5.2 of GA-LTR1 (also enclosed). Previously, it was assumed that the CSB conductivity functions were the same as those shown for the top and bottom reflectors.

Per conversations with Tom Butler, the latest runs were altered to include heat flow printouts for the entire FWCD period. Also, the axial region for which the maximum block-to-block ΔT should be monitored is region 10 (the CSB), rather than region 9 (the bottom half of the lower reflector). Previously, I had assumed that the lower reflector's ΔT's, which were much larger, would have a more significant effect on the local stresses at the top surfaces of the CSB's.

For the 72% power case, the maximum ΔT between adjacent CSB regions is 600°F at t= 240 min, while for the 105% power case it is 602°F at t= 280 min. These differences are much less than those of the previous runs with the lower LTR-1 conductivity values for the CSB.

Please let me know if you have any questions or comments.

Yours truly,

Syd Ball

S. J. Ball, Manager
HTGR Safety Program for NRC/RSR

Enclosure

cc: G. C. Bramblett, GA (w/encl.)
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Ref. 1. General Atomic Co. report on Graphite Design Material Properties, 904434/2 (April 28, 1980), p. 62.

A-1484

GENERAL ATOMIC COMPANY

SJ Ball
12-2-80

TITLE:

GENERIC-GRAPHITE DESIGN MATERIAL PROPERTIES

Document No.	904434	Issue	/2

From G. Bramblett
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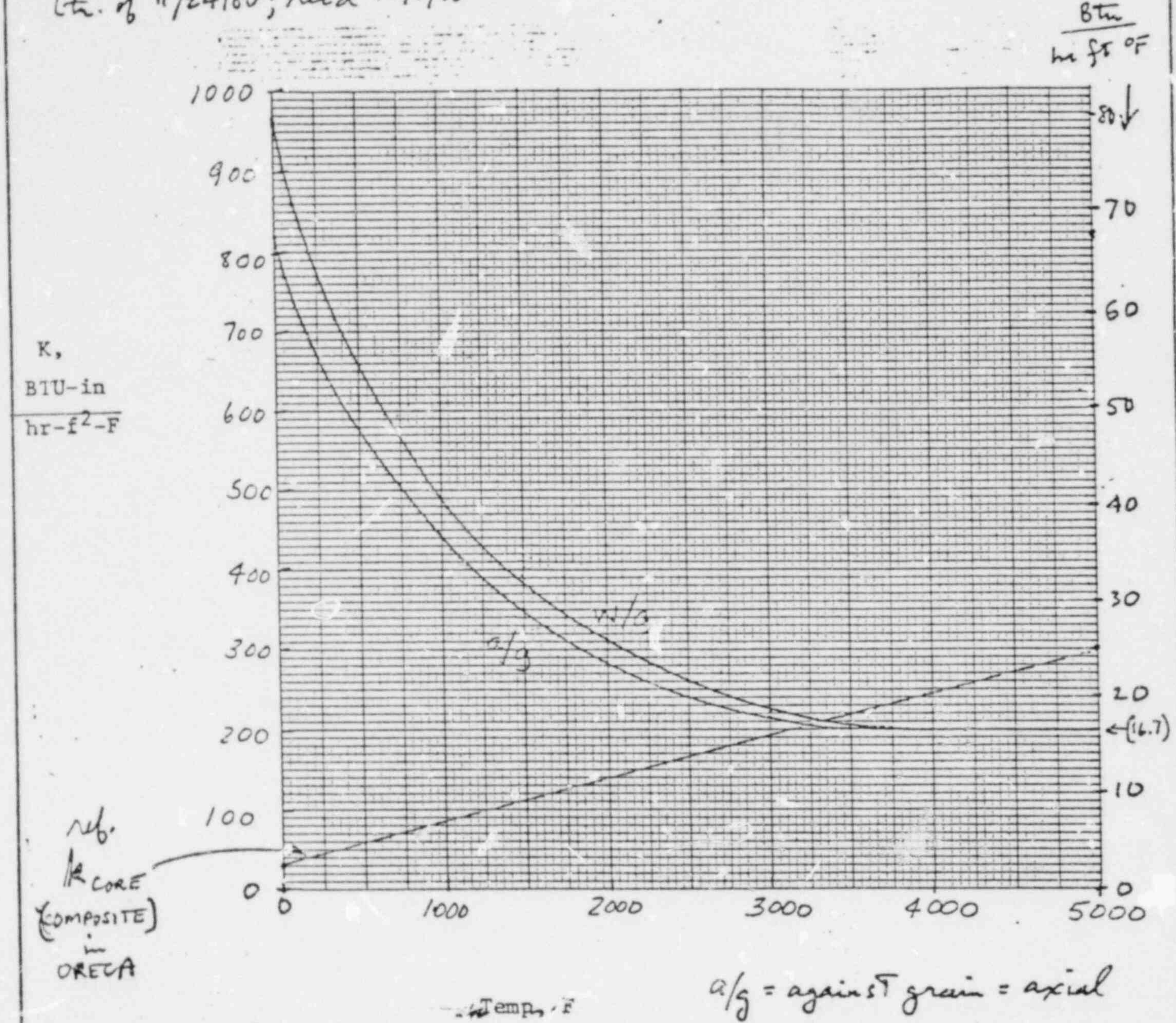

a/g = against grain = axial

w/g = with grain = radial

Figure 4-16. Average PGX Graphite Thermal Conductivity vs. Temperature

• Use for core support block k

$k_{RADIAL} \approx 16.7 + 62.5\, e^{-T/1030}$, BTU/HR·FT·°F ; T = °F

$k_{AXIAL} \approx 16.7 + 50.8\, e^{-T/1100}$ " "

Notations in this column indicate where changes have been made

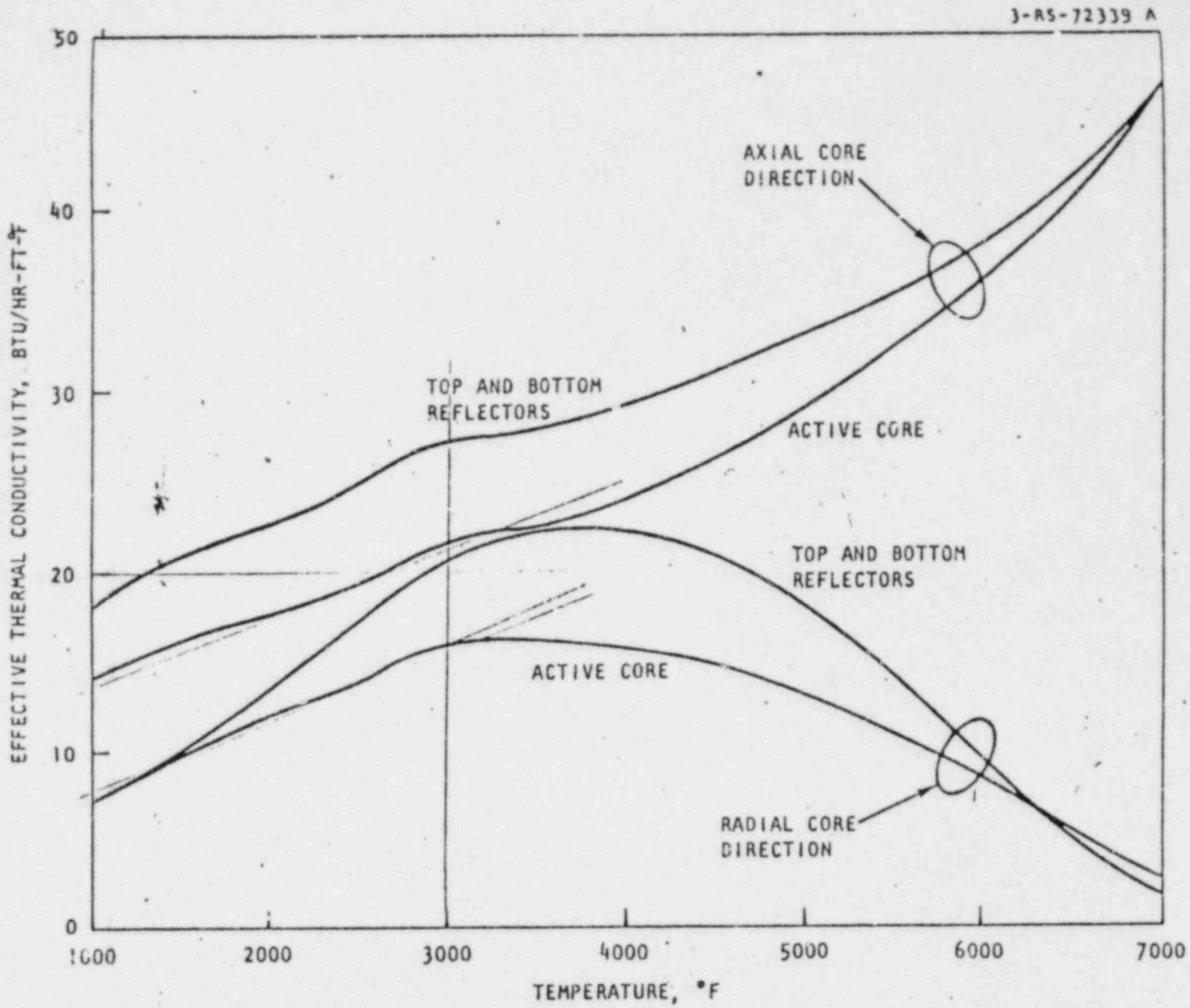

Fig. 5.2. Effective thermal conductivity for a homogenized core

GA - LTR - 1

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HTGR PROGRAM TECHNICAL NOTE

HTGR SAFETY RESEARCH AT THE LOS ALAMOS NATIONAL LABORATORY

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HISTORY REPORT DRAFT

TITLE PAGE

March 1981

ORNL/NUREG/TM-

SUMMARY OF ORNL WORK ON NRC-SPONSORED HTGR SAFETY RESEARCH, JULY 1974 - SEPTEMBER 1980

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PRELIMINARY DRAFT

SUMMARY REVIEW

OF

WORK BY BNL-HTGR SAFETY DIVISION

FOR

ADVANCED SAFETY TECHNOLOGY RESEARCH

OF

NRC

NOTE: This draft is issued for the purposes of review, editing and to elicit constructive questions and comments on its content and organization. It may contain undetected errors, and should be considered in this form as a working paper.

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Experiments will also be carried out to assess how mechanical strength and ductility can be affected by simulated thermal and helium chemistry transients during a hypothetical accident scenario. Fatigue and creep tests initiated under prototypic HTGR conditinos will be interrupted by higher-temperature thermal spikes lasting for several days in order to quantify the losses in fatigue and creep failure times. No work to date has been carried out in this area. Similarly, coolant chemistry transients in which water is injected into the mechanical test system will be performed and the influence on failure rates will be determined.

Additional mechanical tests on alternate structural alloys will also be initiated in FY 1981.

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4. With the extremely low probability of such events further work in this area does not appear warranted at this time, but could be resumed if required by a future shift in emphasis.

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is incurred as the hotter channels will tend to get less flow, thereby becoming even hotter, and getting even less coolant flow, etc.

Figure _.1 shows the flow rates and the temperatures about midstream in a low flow vs. a high flow channel, for three computations, using different gas temperatures for evaluation of average gas densities per node. In the first case the solid block temperature of the node is being used as first approximation for the gas temperature. In the second case an arithmetic average gas temperature is used. A logarithmic average, representing an axial integration with the ORECA assumption of constant graphite block temperature per node is being used in the third case.

As expected, whichever density is used does not make any difference in the high flow channel. However, in the low flow channel average core temperature differences of about 600°F are observed, as the low flow channel will receive less flow, and heat up more over the time period from 100 to 250 minutes. This strong sensitivity of the resulting output temperatures to a relatively minor assumption in the analysis shows that under such low flow conditions the flow redistribution and ultimate core temperatures can be extremely sensitive to minor variations in design and operating parameters such as the flow orificing. An accurate analysis of such situations would require more refined modelling of the specific situation.

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Stearns-Roger
ENGINEERING CORPORATION

March 10, 1981

PUBLIC SERVICE COMPANY OF COLORADO
P.O. Box 840
Denver, Colorado 80201

Attention: Mr. F. Swart
Manager, Nuclear Projects

Reference: Fort St. Vrain Nuclear Generating Station
Unit No. 1
PSCC Document 111942
Stearns-Roger Project C-22240

Dear Mr. Swart:

Pertaining to our conversation today, I have had discussions with several individuals within our organization and cannot substantiate that any report on Fort St. Vrain has been prepared. In addition, it is not our practice to number our reports, and so the Stearns-Roger Report #308 requested by the PUC does not exist.

Please do not hesitate to call if we need to do anything more to eliminate this rumor.

Very truly yours,

STEARNS-ROGER ENGINEERING CORPORATION

J. J. Donovan

J. J. Donovan
Project Manager

JJD:nc

cc: Bill Fitzmorris, PSCC
FLWeigand/LFisher
LMMcBride, PSCC
JJDonovan

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