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HEAT TRANSFER EXPERIMENTS SIMULATING A FAILURE OF THE INLET PIPING TO A K REACTOR PROCESS TUBE

E. D. Waters and D. E. Fitzsimmons

January 20, 1961

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HEAT TRANSFER EXPERIMENTS SIMULATING A FAILURE OF THE INLET PIPING TO A K REACTOR PROCESS TUBE

INTRODUCTION

Reported herein are the results of laboratory heat transfer experiments. These experiments were conducted to investigate fuel element temperatures which could result from coolant flow loss following a failure of the inlet piping to a process tube at a K reactor.

The failure of the inlet coolant piping between the front header and the process tube on a reactor would cut off the flow of cooling water to the fuel elements but should immediately initiate a reactor scram by causing a low trip on the Panellit pressure monitor. However, the reactor power reduction would not be immediate nor absolute and would be dictated by the time required to insert the emergency control rods (VSR's) and by post-scram delayed fission and fission product decay heating. The only means of heat removal from the affected tube and fuel elements during the post-scram period would be by reverse flow of hot water from the rear header. The objective of the subject experiments was to determine what rear header pressures would be required to achieve adequate cooling of a K reactor fuel assembly during such a post-scram period. Such information is of value in updating of reactor hazards evaluation reports.

Experimental studies were previously reported concerning failure of a front hydraulic fitting on a C reactor 'Operational-charge-discharge' geometry.[1,2,3]* Preliminary experiments were also conducted for a standard K reactor coolant inlet geometry,[4] but failure of the test section caused postponement of the program. The previous studies did indicate appreciable differences in the rear header pressures required for adequate post-scram cooling of the two assemblies and thus the C reactor data could not be applied to K reactor.

The study reported here was carried out by the Thermal Hydraulic Operation in the 189-D Heat Transfer Laboratory.

SUMMARY

Electrical resistance heating of a metal test section was used to simulate a 38 piece charge of K-III I&E fuel elements in a standard K process tube and hydraulic fitting assembly. Failure of the inlet piping to a single process tube was simulated at equilibrium tube powers of 750 to 1800 KW with rear header pressures of 15 to 90 psig. Three seconds after the simulated failure, the power input to the test section was reduced in accordance with a 500 ih scram and the reverse flow rate, rear header pressure and the temperatures from 12 thermocouples on the heater rod were recorded during the transient

*Numbers in parentheses refer to entries in Bibliography, page 9.

conditions. The effects of scram delay time and rear header water temperature were briefly investigated.

Experimental data from the transient experiments are shown on Figure 1. These data relate the maximum observed heater rod (simulated fuel) temperature to the rear header pressure and the steady state tube power prior to an inlet piping failure. A comparison is made in this report between the experimental conditions and typical reactor conditions. This comparison includes as an appendix, a discussion of the effect of the specific heat of the fuel on the fuel temperature during transient experiments. This comparison indicates that reactor fuel elements would experience a temperature rise of about 1.30 to 1.4 times that which was experienced with the experimental equipment. Thus, a maximum temperature of 960 to 1010 F for the heater rod (as on Figure 1) would correspond to the aluminum melting point (1220 F) for the same rear header pressure on the reactors.

The experimental results indicate that K-reactor central zone tubes would be subjected to fuel jacket melting upon failure of individual tube inlet piping with present tube powers of about 1500 KW and with present rear header pressures of 15 to 60 psig.

Data were also obtained under conditions of constant power level and steady reverse flow from the rear header through the tube and out a 'failed' inlet connector to atmosphere. From these experiments, steady state hydraulic demand curves have been drawn as shown in Figure 2. These data compare well with steady state demand curves for normal forward flow in a K reactor assembly and thus indicate that the present studies may be applicable to situations where a sudden reduction in front header pressure occurs.

DISCUSSION

Events Involved in Piping Failure Incident

The sequence of events during a single tube inlet piping failure incident is postulated as follows: The water pressure at the tube inlet drops very rapidly to atmospheric pressure and a low trip is experienced by the Panellit pressure monitor. The cold water flow from front header through the tube ceases and hot water begins to move from the rear header into the tube (reverse flow) at a very low rate under the influence of the rear header pressure. This all occurs within one second. During this time, very vigorous boiling occurs throughout the coolant channel as the system is 'de-pressurized' from its original condition and as flow stagnation and reversal takes place. Large quantities of vapor form and begin to move toward the failed 'inlet' piping.

The reactor primary safety circuit should have been actuated by the Panellit low trip. But by the time the vertical safety rods (VSR's) are wholly effective (about 2 seconds after piping failure), the surface of the fuel pieces will be blanketed by a layer of steam and the surface temperature may well have reached 500 to 700 F. The heat generation rate decreases rapidly at first after the VSR's become effective but the reverse flow rate remains low (1 to 4 gpm) for many seconds and the fuel temperature continues to rise because of insufficient heat removal. As the seconds pass, the heat generation rate in the process tube continues to fall, as does the hydraulic demand pressure which would be necessary to achieve single phase (liquid) cooling in the assembly. Only when the maximum point of the hydraulic demand curve (typified on Figure 2) falls below the available rear header pressure can the assembly begin to recover from the film boiling conditions. This is referred to as having excess header pressure.

The time required to reach a condition of excess header pressure is dependent upon the available rear header pressure, the initial heat generation rate and the heat generation decay curve. However, even after excess header pressure conditions are established it takes time before cooling conditions will change from film boiling to all liquid cooling throughout the tube. This is because there is considerable stored heat which must be removed by the water as it progresses along the tube in overcoming film boiling conditions. During this time the fuel temperature will continue to increase in the regions where film boiling still prevails. Therefore, it will actually be some seconds after reaching excess header pressure that the fuel temperature reaches a peak value and begins to decrease. It was the purpose of these experiments to simulate this sequence of events and to investigate the relationship between maximum fuel temperature and the variables--rear header pressure, initial heat generation rate, and mode of heat generation rate change.

Experimental Apparatus and Procedure

The experiments were conducted with the 189-D Heat Transfer Apparatus with nuclear heat being simulated by d.c. electrical resistance heating in a metal 'rod'. The heater rod dimensions (1.447 inch O.D. by 0.391 inch I.D.) were selected to be approximately midway between those of K-III-N and K-III-E fuel elements but included a 0.010 inch reduction of O.D. as an allowance for the "Heresite" electrical insulation coating on the inside of the standard K process tube. The heater rod had a stepped cosine power distribution with a peak power to average power ratio of 1.43. Six hold-down pins were inserted through the top of the process tube at approximately equally spaced intervals to restrain upward bowing or "cocking" of the heater rod in the process tube. While these hold-down pins do aid in preventing contact of the heater rod with the top of the process tube, their use does not assure that such contact is prevented and certainly does not prevent all possibilities of eccentricity of the heater rod and process tube at locations between the hold-down pins.

Twelve thermocouples were imbedded about 1/16 inch below the top outside surface of the heater rod at approximately equal space intervals along the rod. These thermocouples give temperature readings which are related to the surface temperature but are slightly higher than the surface temperature. At steady state conditions with 1800 KW heat input, the temperature at 1/16 inch below the surface may be up to 150 F higher than the surface temperature. But at about 20 seconds after scram, the temperature distribution through the heater rod should have changed such that the imbedded thermocouple would indicate a maximum of 30 F high. Comparison of steady state thermocouple readings with predicted non-boiling surface temperatures indicated that the thermocouples were less than 1/16 inch below the surface of the heater rod.

The coolant inlet and outlet fittings were of standard K reactor type. Failure of an inlet hydraulic fitting was simulated in the following manner. Normal operating flow rate and power level conditions were established to give an outlet temperature of 125 C at steady state conditions. Rear header pressure was adjusted and maintained constant by a pressure recorder-controller operating in conjunction with a small pump connected to the rear header. Two air operated valves were then actuated simultaneously to stop the supply of water to the front header and open the front pigtail (at the header end) to a 3 inch pipe which in turn discharged at atmospheric pressure to drain. Three seconds after valve actuation, the power reduction was started and carried out in simulation of a reactor scram. (Two different power decay curves were used for each tube power, except 750 KW. These two decay curves will be discussed later). Dependent variables (temperatures, flow rate and tube inlet pressure) were recorded as the power decay was carried out and until adequate heater rod cooling was re-established as indicated by decreasing rod thermocouple readings.

The steady state hydraulic demand data were obtained with flow from the rear header through the tube and out the front pigtail which discharged at atmospheric pressure. During these steady state runs, the water entering the tube from the rear header was cold (66 F).

Results

The results of the transient experiments simulating a front face fitting failure are presented on Figure 1 as 'Maximum Heater Rod Surface Temperature During the Transient' versus 'Rear Header Pressure' for various initial tube power levels. The results of the steady state experiments are shown on Figure 2 in the form of 'Rear Header Pressure' versus 'Flow Rate' for various constant low tube powers. The following discussion will serve to compare the experimental conditions with actual reactor conditions and thus aid in applying the experimental results to evaluation of actual reactor hazards.

Steady State Data

The steady state hydraulic demand data of Figure 2 have little direct application to reactor operation. Their main use is to aid in planning of transient experiments and in analysis of transient data.

By use of these steady state data with a selected rear header pressure and power decay curve, one can predict the time (power level) after pigtail failure at which a condition of excess rear header pressure would be attained. For example, on Figure 2 a 38 psig rear header pressure would correspond to the peak demand pressure for a 150 KW tube power. Therefore, 38 psig would be termed the 'excess header pressure' for this instantaneous tube power. If one began with a 1000 KW tube power and power decay occurred as per a 500 ih scram, it would be about 18 seconds after scram that the instantaneous power was 150 KW. This would be the time required to reach excess header pressure. Any time later than this, conditions should be progressing from film boiling toward conditions of single-phase, forced convection heat transfer.

Comparison of the 'predicted time for excess header pressure' with the experimentally observed sequence of events showed the following: (1) Maximum heater rod surface temperatures occurred at about twice the time interval predicted to reach excess header pressure conditions. (2) Recovery to single phase convective heat transfer conditions required about 3 to 8 times the time interval predicted to reach excess header pressure. It depended greatly on the rear header pressure and initial tube power conditions.

Comparison of the hydraulic demand curves on Figure 2 with those given in HW-66123[5] shows that the steady state demand pressures are quite similar for the cases of forward and reverse flow. This would indicate the possible application of the transient data from these studies to occurrences of sudden front header pressure losses or reductions.

Experimental Heat Input

The transient data of Figure 1 show two curves for each initial tube power. The difference between the two curves results from using different power decay curves in the programmed power reduction during the simulated scram, as follows: The lower curve of each case represents a 500 ih VSR rod strength with the experimental heat input rate equal to the instantaneous nuclear heat generation rate in a reactor lattice (both in fuel and graphite) as given by Figure 1 of HW-33870[6]. For the upper curve of each case, the experimental heat input rate equalled the instantaneous nuclear heat generation rate (fuel and graphite) plus five per cent (5%) of the initial tube power. The additional 5 per cent of initial tube power is an allowance for sensible heat transfer from the graphite to the coolant. Generating extra heat in the heater rod is presently the only way of allowing for graphite sensible heat in the experimental apparatus. Such a procedure should result in higher-than-prototypical heater rod temperatures[7] but does serve to set an upper limit on the effect of the graphite sensible heat contribution.

If 500 ih VSR strength is too low, then the lower curve of each case would be approximately equivalent to an 800 ih scram with an extra 5 per cent for graphite sensible heat. Thus the lower curve for each case should approximate an upper limit on heater rod temperatures for an 800 ih scram or a lower limit on heater rod temperatures for a 500 ih scram.

All data points on Figure 1 are for a 3 second delay between simulated pigtail failure and the beginning of power reduction. The effect of delay time on maximum temperature was cursorily determined for the 1000 KW case. It was found that the maximum observed temperature was about 100 F lower when the delay time was decreased by one second. This corresponds quite closely to the difference in the heater rod temperature rise which one would calculate by assuming no heat transfer to the water during this period (i.e. 89 F). Thus, as an approximation, one might assume approximately 0.090 F/KW-sec as the change in maximum temperature of the heater rod for changes in the delay time between pigtail failure and initiation of power decay.

Effect of Rear Header Water Temperature

The transient experiment data points shown on Figure 1 were all obtained while using a rear header water temperature of 125 C, except for two points. These two points were obtained with a rear header temperature of 86 C for the 1250 KW nuclear heat only case (lower 1250 KW curve). Comparing data points for the two different temperature conditions, one finds that rear header temperature does affect the maximum surface temperature which is attained during a pigtail failure occurrence, but the effect is equal to or less than difference in rear header temperatures. Since the range of rear header water temperatures is small on the reactor, this effect would be of little importance. From this standpoint, the data of Figure 1 are conservative in application to reactor hazard studies.

Heat Capacity of Experimental Heater Rod

One of the biggest questions concerning the application of the transient data of Figure 1 to reactor hazard evaluation is the effect of the difference in heat capacity between the experimental heater rod and actual reactor fuel elements. The heat capacity of the heater rod used in these experiments was 0.53 Btu/°F per foot of length. But the effective heat capacity of a canned uranium fuel element of the K-III size would be $\sim$ 0.36 Btu/°F per foot of length or perhaps even lower. Thus the experimental heater rod was capable of storing at least 1.48 times as much heat for a given temperature rise as the uranium charge which it simulated.

The problem of the effect of heat storage capacity of the fuel on fuel temperature during transient experiments was investigated analytically. This is discussed as Appendix A, page 10 of this report. Although there were many assumptions involved, it was concluded that it would be reasonable to say that the surface temperature rise which would be experienced by uranium fuel elements would be 1.3 to 1.4 times that which was observed with the experimental heater rod. If one uses 1220 F (aluminum melting point) as the maximum allowable fuel element temperature and 300 F as the surface temperature before the piping failure, then experimentally observed temperatures of 960 F to 1010 F on Figure 1 would correspond to a minimum acceptable condition of adequate cooling for the reactor.

[redacted]

General Observations

In lieu of presenting flow rate data for the transient experiments, the following general comments are offered regarding flow rates. For the combinations of rear header pressures and original tube powers given on Figure 1, the flow rate from the rear header through the tube ranged from about 0.5 to 4.0 gpm during the interval from simulated inlet piping failure to complete recovery from film boiling conditions. Following recovery from film boiling conditions, the flow rate rose to the value indicated on Figure 2 for single phase (isothermal) flow with the appropriate rear header pressure.

During the transient experiments, the maximum heater rod temperature always occurred in the region of highest heat flux. The specific location of the hot spot varied along this 8 foot length of high heat flux in the center of the rod but no comparably high temperatures were ever detected upstream or downstream of this section. The 'normal upstream' (front face) portion of the heater rod experienced temperatures which were generally slightly above those of the 'normal downstream' portion. Superheated steam discharge was not encountered in any of the transient tests. Steam discharge temperatures were generally 275 F or less in the front nozzle barrel.

The fact that no high temperatures were detected on the 'normal downstream' portion of the heater rod indicates the absence of flow channeling which was suspected in the tests reported in HW-61849[4]. During the steady reverse flow experiments of the present study, several of the hold-down pins were loosened to determine if the top of the heater rod was covered with water at low flow conditions. It was found that even a 0.5 gpm flow rate was sufficient to completely cover the heater rod in the process tube.

For the transients experiments, a comparison was made between the measured temperature rise and the calculated temperature rise of the experimental heater rod based on the integrated heat input since simulated inlet piping failure. It was assumed that no heat was transferred from the rod to the water during this time. Based on the heat input during the time interval necessary to reach excess header pressure, as discussed previously, it was found that the measured maximum temperature rise of the heater rod was only 0.6 to 0.9 of the calculated temperature rise. While this does not serve to accurately predict the maximum temperature during such a transient condition as inlet piping failure, it may serve as a useful extrapolation or approximation tool.

E. D. Waters

D. E. Fitzsimmons

Thermal Hydraulics Operation
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APPENDIX A

ANALYTICAL APPROXIMATION OF THE EFFECT OF FUEL HEAT CAPACITY ON FUEL TEMPERATURE DURING FLOW STOPPAGE TRANSIENTS

Calculations were performed to estimate the effect of 'fuel' heat storage capacity on the maximum temperature which might occur during a severe transient coolant flow reduction. Several simplifying assumptions were used in conjunction with a heat balance equation. The results of these analytical approximations indicate that the meager heat transfer rates which are achieved during such a flow transient as an inlet piping failure are sufficient to reduce the effect of 'fuel' heat storage capacity by about one third. That is, if the temperature rise ratio for two different fuel materials was predicted to be 1.50 strictly on the basis of heat storage capacity, then the actual temperature rise ratio for the two materials under transient conditions similar to those described would be only about 1.33. (If heat storage capacity had no effect on temperature the temperature rise ratio would be 1.0). A comparison between a calculated temperature rise curve and an experimental temperature rise curve for the experimental heater rod indicates that the analytical assumptions which were used are reasonable.

Consider the following analysis. For any segment of an experimental heater rod or an actual fuel element in which heat is generated, an energy balance can be written in differential form as follows for a unit volume of the material:

(heat generation rate) = (heat transfer rate) + (heat storage rate)

$$q = UA(T_s - T_c) + \rho C_p \frac{d(T_{metal})}{d\theta} \qquad \text{(A)}$$

where q = heat generation rate per unit volume, $\frac{\text{Btu}}{\text{hr}\cdot\text{ft}^3}$

U = overall coefficient of heat transfer, $\frac{\text{Btu}}{\text{hr}\cdot\text{ft}^2\cdot\text{F}}$

A = heat transfer area per unit volume, ft^2/ft^3

T_s = surface temperature of metal, °F

T_c = bulk coolant temperature, °F

C_p = specific heat of metal, Btu/lb-°F

ρ = density of metal, lb/ft^3

$T_m = T_{metal}$ = 'average' metal temperature, °F

θ = time, hrs

[redacted]

Equation (A) can be simplified by assuming (for the time being) that the thermal conductivity of the 'fuel piece' material is very large such that the metal surface temperature is equal to the 'average' metal temperature. Also, it is convenient to reference all temperatures to the coolant temperature which is nearly constant during the boiling conditions associated with an inlet piping failure transient. Then, equation (A) becomes

$$q = UAT_m + \rho C_p \frac{dT_m}{d\theta} \qquad \text{(B)}$$

or re-arranging

$$\frac{dT_m}{d\theta} + bT_m = c \qquad \text{(C)}$$

where

$$b = \frac{UA}{C_p \rho} \quad \text{(assumed to be constant)}$$

$$c = \frac{q}{C_p \rho}$$

If the heat generation rate, q, is a constant, then equation (C) can be integrated to obtain

$$T_m = (T_{m_o} - c/b)\, e^{-b\theta} - c/b \qquad \text{(D)}$$

where T_{m_o} is the 'average' metal temperature at the beginning of the transient condition. But if the heat generation rate changes with time in an exponential manner such that

$$q = q_o\, e^{a\theta} \qquad \text{(E)}$$

then equation (C) will integrate to

$$T_m = \left(T_{m_o} - \frac{c}{a+b}\right)e^{-b\theta} + \left(\frac{c}{a+b}\right) e^{a\theta} \qquad \text{(F)}$$

The power decay period of the transient can be broken down into several segments to allow use of different values of a in equations (E) and (F) to fit the desired heat generation shutdown curve.

Notice that equations (D) and (F) contain a heat transfer coefficient in the term b which was assumed to be constant. Actually, the heat transfer coefficient, U, will vary during such transient conditions as an inlet piping failure. But since U cannot be accurately predicted or determined during such transient film boiling conditions, there was little choice but to assume 'reasonable' values for U and to use the same values of U for the experimental heater rod and for uranium fuel elements in comparing such cases.

[redacted]

The above integrations assume constant values of density, ρ, and specific heat, C_p, during the transient condition. Since our interest is in a comparison between different cases rather than absolute values and since the variation of ρ and C_p is of the same order of magnitude for the materials under consideration, then this assumption appears justified.

'Average' metal temperatures were calculated according to equations (D) and (F) for three different cases: (1) An experimental heater rod of 70% copper - 30% nickel with dimensions as given in this report. Heat storage capacity = 0.532 Btu/°F per foot of length. (2) Canned uranium fuel elements including heat storage capacity of uranium and aluminum jacket (excluding end caps) with 40 mil jacket thickness and same OD and ID as K-III I&E fuel elements. Heat storage capacity = 0.359 Btu/°F per foot of length. (3) Uranium fuel pieces on a 'bare' basis--that is, the uranium had the same dimensions as the uranium in case (2) but there was no allowance for the heat storage capacity of any aluminum. Heat storage capacity = 0.298 Btu/°F per foot of length.

The difference between metal surface temperature and 'average' metal temperature can be calculated for steady state heat transfer conditions as a function of heat generation rate and the thermal conductivity of the material. A calculated value of this temperature difference should be valid for our time zero (which is steady state) and should be fairly valid for any time after about 10 seconds of the transient experiment since the calculated difference is then small and changing only slowly. The 'average' metal temperatures calculated by relations (D) and (F) were adjusted in this manner to show surface temperature versus time on Figure 3.

The ratio of heat storage capacity of the experimental heater rod to that of 'bare uranium' is $\frac{0.532}{0.298} = 1.78$, so that the 'average' temperature rise of the uranium should be 1.78 times that of the heater rod for the same heat input with no heat transfer. However, the curves on Figure 3 show that the low heat transfer rate, which was assumed, was sufficient to lower the temperature rise in the 'bare uranium' to only 1.37 times that in the experimental heater rod. Similarly, the temperature rise for the 'canned uranium' should be about $\frac{0.532}{0.359} = 1.48$ times that for the heater rod if no heat transfer to the coolant. But the calculated curves show that the meager heat transfer is sufficient to keep the temperature rise of the 'canned uranium' equal 1.25 times that of the heater rod. Thus, the low heat transfer rates which were assumed for this case were sufficient to reduce the heat storage effect to about one half of what it would be if no heat were transferred to the coolant during the piping failure incident. Assumption of lower heat transfer coefficients would cause the temperature rise ratio to approach the heat storage capacity ratio. This is shown by the following summary of calculations involving various assumed heat transfer coefficients over different times of the transients.

[redacted]

Heat Transfer Coefficients*	Maximum Heater Rod Surface Temp.	Temperature rise Ratio	
Btu/hr-ft^2 °F	°F	Canned Uranium to Heater Rod	Bare Uranium to Heater Rod
400; 100; 60	834	1.25	1.37
400; 100; 30	930	1.29	1.44
400; 100; 15	1005	1.32	1.49
0	--	1.48	1.78

* 400 applies for the time 0 to 3 seconds, 100 applies for the time 3 to 4 seconds, and the third value applies for the time of 4 seconds and later.

The question remained 'Were the assumed heat transfer coefficients reasonable?' Experimental data points from a typical test run were plotted on Figure 3. These data compare quite well with the calculated temperature curve for the experimental heater rod. This is somewhat fortuitous because a test run with a different rear header pressure would result in a different temperature rise in the heater rod. But this comparison does show that heat transfer coefficients in the range 60 to 400 Btu/hr-ft^2 are realistic.

It is estimated that actual fuel elements in a reactor process tube would behave some place between the 'bare uranium' and the 'canned uranium' cases of Figure 3. It is then concluded that the surface temperature rise of K reactor fuel elements would be 1.30 to 1.40 times that which was observed in the subject experiments and was plotted on Figure 1.

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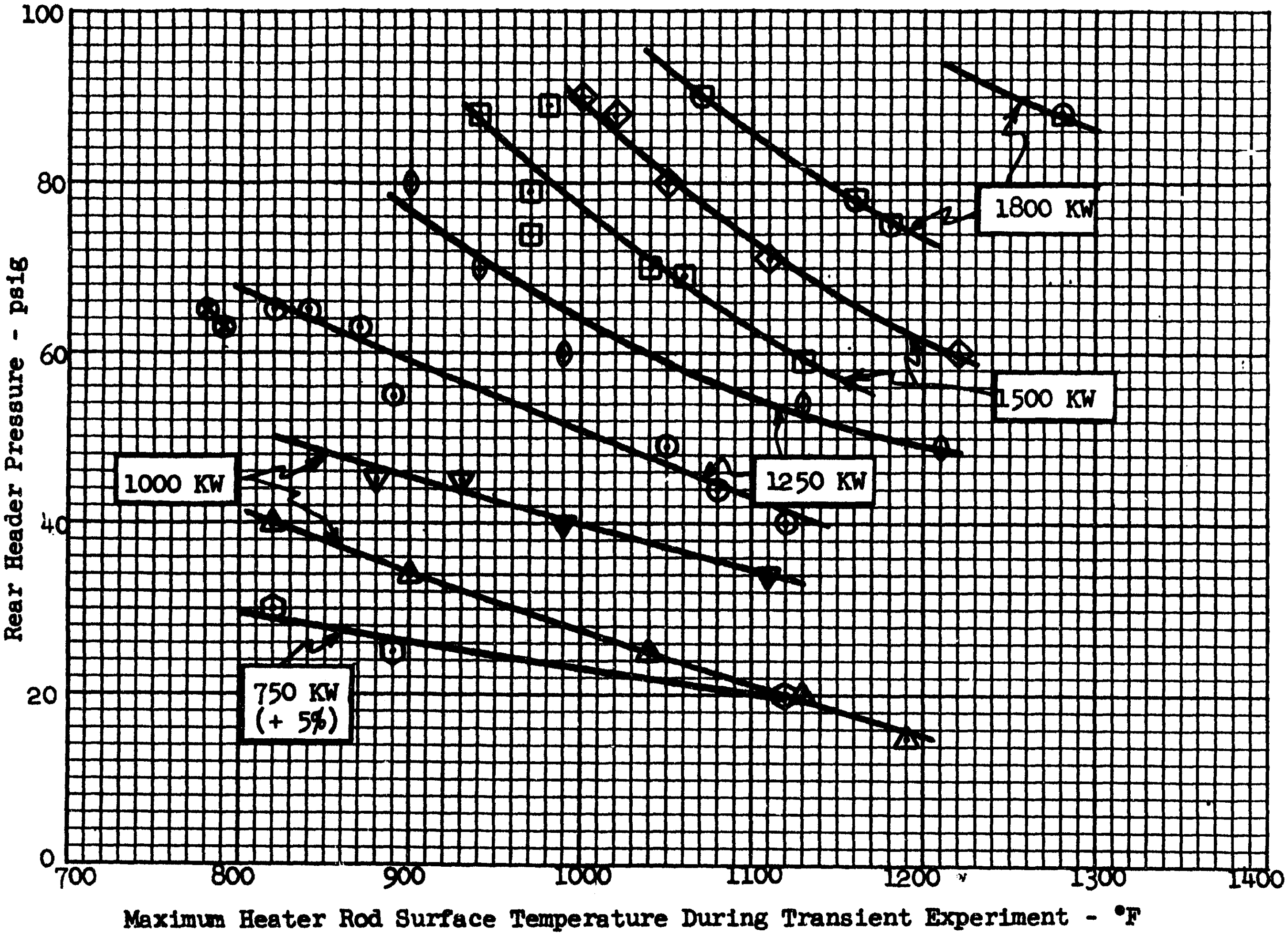

Figure 1. Results of Front Connector Failure Experiments.

Basis: K-Reactor Mock-up with front pigtail open to atmosphere at header end. Temperature of water from rear header = 125 C, except for symbols ⊗ with water temperature = 86 C. Simulated 38 piece K-III I&E fuel charge. See discussion page 6 regarding upper and lower curves for each initial tube powers. Rod surface temperature = 300 F before transient began.

NOTE: Temperature rise of reactor fuel will be about 1.3 times that of heater rod. See discussion, page 7, and Appendix A, page 10.

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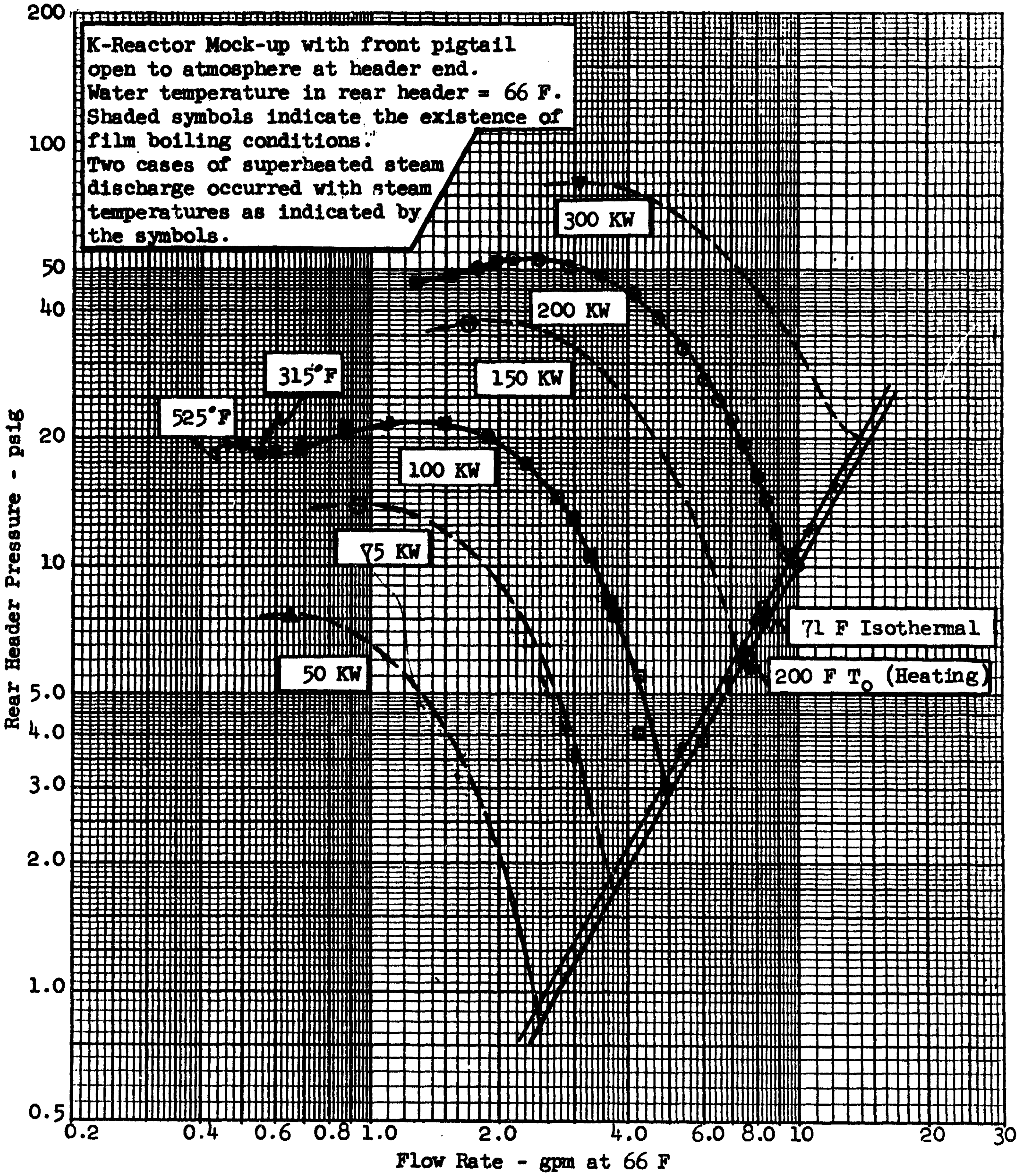

Figure 2. Reverse Flow 'Boiling" Hydraulic Demand Data.

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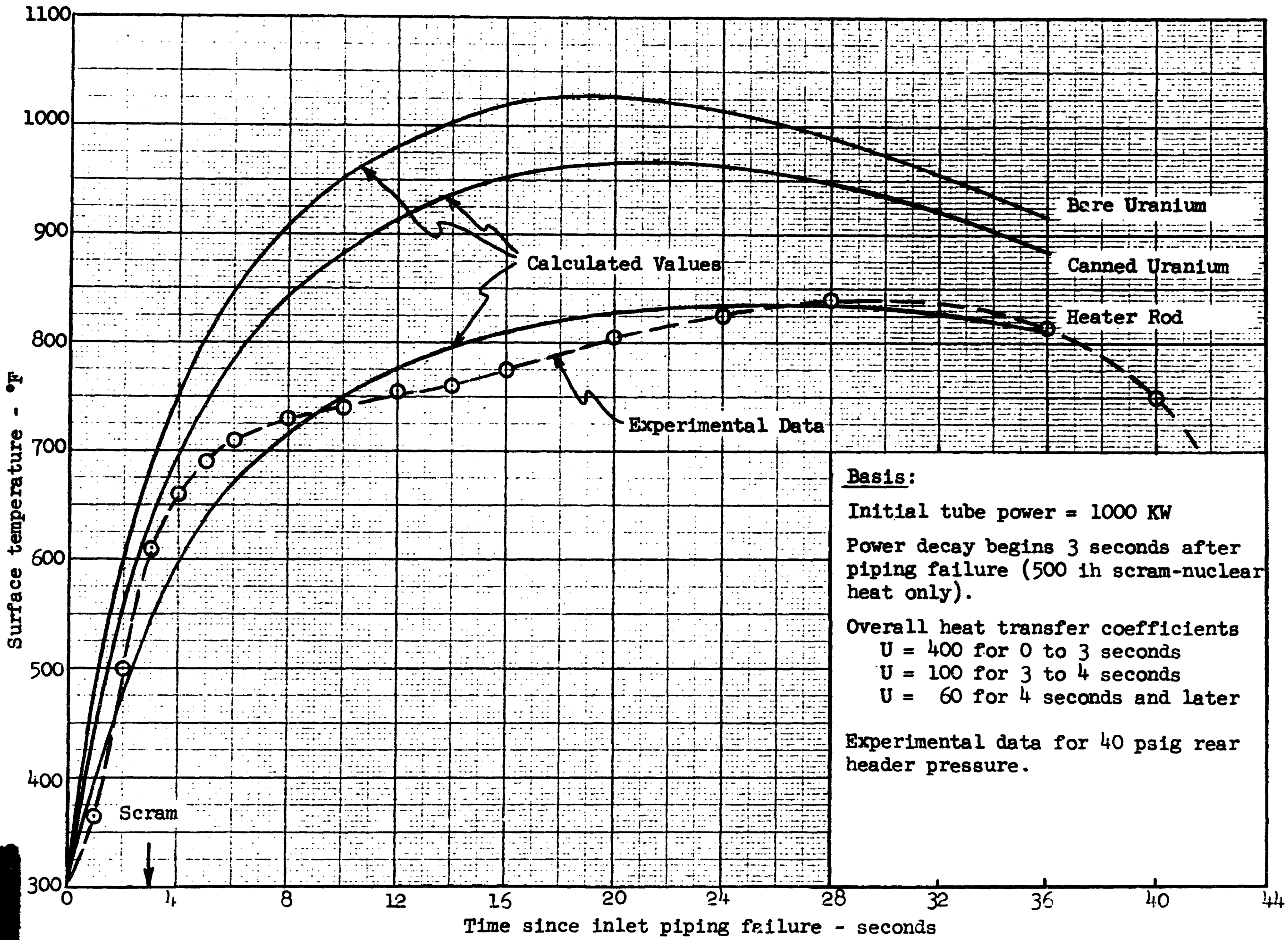

Figure 3. Comparison of Different 'Fuel' Materials During a Flow Transient.

[redacted]