AIIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910

301/587-8202

Centimeter

Inches

MANUFACTURED TO AIIM STANDARDS
BY APPLIED IMAGE, INC.

1 of 1

HANFORD CODE

DOCUMENT NO. HW-70769

[redacted]
(CL[redacted]SIFICATION)

DECLASSIFIED

SERIES AND COPY NO. [redacted]

GENERAL ELECTRIC

HANFORD ATOMIC PRODUCTS OPERATION - RICHLAND, WASHINGTON

DATE August 16, 1961

[X] REST[redacted]DATA

THIS DOCU[redacted]AINS RESTRICTED D[redacted] DEFIN[redacted] ATOMIC ENERGY [redacted]. [redacted]MITTAL OR THE DI[redacted] ITS [redacted]TS IN ANY MANNER[redacted]UTHORIZED [redacted]ERSON IS PROHIBITED.

TITLE

HEAT TRANSFER EXPERIMENTS SIMULATING A FAILURE OF THE INLET PIPING TO A BDF REACTOR PROCESS TU

[ ] OTHER OFFICIAL CLASSIFIED INFORMATION

THIS MATERIAL CONTAINS INFORMATION AFFECTING THE NATIONAL DEFENSE OF THE UNITED STATES WITHIN THE MEANING OF THE ESPIONAGE LAWS, TITLE 18, U. S. C., SECS. 793 AND 794, THE TRANSMISSION OR REVELATION OF WHICH IN ANY MANNER TO AN UNAUTHORIZED PERSON IS PROHIBITED BY LAW.

AUTHOR

E. D. Waters
M. R. Kreiter

ISSUING FILE

CIRCULA
RECEIVED
AUG

THIS DOCUMENT MUST [redacted]EFT UNATTENDED OR WHERE [redacted]ZED PERSON MAY HAVE ACCES[redacted] TO IT. WHEN NO[redacted]MUST BE STORED IN AN [redacted]CKED REPOSITORY WITHIN AN [redacted] GUARDE[redacted]ILE IT IS YOUR POSSESSIO[redacted]IL YOU HAVE OBTAINED A SIGNED [redacted]ROM [redacted] FILES, IT IS YOUR RESPON[redacted]O KEEP IT AND ITS CONTENTS WI[redacted]IMITS OF [redacted] PROJECT AND FROM ANY UNAU[redacted]ERSON. ITS TRANSMITTAL TO[redacted] AT YOUR PLACE OF RESIDENCE IS PROHIBIT[redacted] NOT TO BE DUPLICATED. I[redacted] COPIES ARE REQUIRED, OBTAIN THEM FROM T[redacted]D ISSUING FILE. ALL PERSONS RE[redacted]HIS DOCUMENT ARE REQUESTED TO SIGN IN THE SPAC[redacted]VIDED BELOW.

ROUTE TO: [redacted] SIGNATURE AND DATE

[redacted]

MASTER

D[redacted]BUTION OF THIS DOCUMENT IS UNLIMITED

DECLASSIFIED

DECLASSIFIED

[redacted]

HW-70769

This document classified by

J M Batch

J. M. Batch

This document consists of 11 pages. No. 12 of 31 copies. [redacted]

DISTRIBUTION

1. JD Agar
2. FW Albaugh
3. TW Ambrose
4. JT Baker
5. JW Baker
6. JM Batch
7. JH Brown
8. PA Carlson
9. RG Clough
10. DL Condotta
11. JW Finnigan
12. JC Fox
13. RM Fryar
14. SM Graves
15. AK Hardin
16. H. Harty
17. HW Heacock
18. SS Jones
19. MR Kreiter
20. WR Lewis
21. LT Pedersen
22. RW Reid
23. HG Spencer
24. FW VanWormer
25. AP Vinther
26. ED Waters
27-29. Extra
30. 300 File
31. Record Center

August 16, 1961

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

E. D. Waters and M. R. Kreiter

Classification Cancelled and Changed To

DECLASSIFIED

By Authority of R M Iten

CGPR-2 3-2-94

By J E Savely 3-15-94

Verified By Tok Hanson 3-25-94

[redacted]

[redacted] document contains restricted [redacted] its transmittal
in [redacted]
or the [redacted] manner to
[redacted]ized person is prohibited.

DECLASSIFIED

[redacted]

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

INTRODUCTION

Laboratory heat transfer 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 B, D, F, DR, or H reactor. The results are reported herein.

Failure of the inlet coolant piping between the front header and the process tube on a reactor would stop the normal flow of cooling water to the fuel elements. Such a failure should immediately initiate a reactor shutdown, but the only means of removing the heat released during the post-shutdown period would be by reverse flow of hot water from the rear cross header. The subject experiments were conducted to determine what rear header pressure would be required to achieve adequate cooling of a BDF type reactor fuel assembly following such a piping rupture.

Experimental studies were previously reported concerning failure of inlet piping to a K reactor geometry. (1)* The analytical techniques and experimental procedures used previously were also used in the present experiments. Rather than to repeat much of this information, numerous referrals will be made to reference (1). It is recommended that this reference be consulted if knowledge of these details is desired.

The study reported herein 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 32 piece charge of O-II-N internally and externally cooled fuel elements in a standard BDF process tube and hydraulic fitting assembly. Complete failure of the inlet piping to a single process tube was simulated at equilibrium tube powers of 800, 1200, and 1400 KW. Various constant rear header pressures in the range of 15 to 100 psig were used. Three seconds after the simulated failure, the power input to the test section was reduced in accordance with an 1100 ih scram while transient pressures, temperatures, and flow rates were recorded on high speed recording equipment.

Data from the experiments are shown on Figure 1. These data relate the maximum observed heater rod (simulated fuel) temperature to the rear header pressure and to the steady state tube power prior to an inlet piping failure. The data were obtained for two different power decay curves for each initial tube power condition. These different decay curves represented an approximate upper and lower limit on the effect of the sensible heat release from the reactor graphite stack during the post-shutdown conditions.

[redacted]

Information from Figure 1 is cross-plotted on Figure 2 (solid lines) to show the rear header pressure which is required to prevent fuel jacket melting as a function of the initial tube power. This includes a correction factor for the effect of specific heat capacity differences between the uranium fuel pieces and the experimental heater rod.

The experimental results indicate that the majority of old reactor central zone tubes would not be subjected to fuel jacket melting upon failure of individual tube inlet piping with present tube powers of about 1200 KW and present rear header pressures. These results are contrary to those found for the K reactors (1) which are shown by the dashed lines of Figure 2. The difference between a K reactor assembly and an old reactor assembly, as far as the rear header pressure required to prevent fuel jacket melting, can largely be explained by the difference in the assumed Vertical Safety Rod strength and hence the power decay curve which was used for the different experiments.

DISCUSSION

Experimental Apparatus and Procedure

The equipment and procedures which were used in the present experiments were the same as described in reference (1) for the K reactor experiments, with two exceptions: First, the heater rod for the BDF experiments had internal and external dimensions equivalent to O-II-N I&E fuel pieces. The stepped cosine power distribution resulted in a peak power to average power ratio of 1.39. The heated length was 23.5 feet; Second, the hydraulic piping and fittings used for the BDF experiments were of standard CG-558 front face geometry and standard BDF rear face geometry with a helical (duPont) rear pigtail.

Steady state flow rate and power level conditions were established in the apparatus before each simulation of inlet piping failure. The outlet water temperature under these conditions was established at 110 C for the 800 and 1200 KW runs and 125 C for the 1400 KW runs. These were the temperatures of the reverse flow coolant from the rear header during the transient conditions of a simulated piping failure. As was discussed in reference (1), the effect of reverse flow coolant temperature is quite small.

During each transient experiment, the electrical power input to the heater rod was varied to simulate the heat output from a reactor lattice following a Vertical Safety Rod (VSR) scram. Two different power decay curves were used in obtaining data for each initial tube power condition. For one decay curve, the power at any time, θ, after scram was determined from reference (2), (Figure 1), as the total theoretical nuclear heat generation (uranium plus graphite) at that time. For the other power decay curve, the electrical power input at time, θ, was equal to the total theoretical nuclear heat generation (uranium plus graphite) at that time plus 5 per cent of the tube power at time $\theta = 0$, (i.e. 5 per cent of the initial steady state tube power). In each case, an 1100 i.h. VSR strength was assumed.

Events Involved in Piping Failure Incident

The sequence of events during a single tube inlet piping failure incident is described in reference (1). This sequence should not vary significantly for a change in inlet piping geometry. The effect of such a piping failure on the principal process variables is shown schematically in Figure 3. The reverse flow of hot water from the rear header begins almost immediately and hot fluid (steam and water) is discharged from the front nozzle of the process tube. The fuel surface temperature rises very rapidly during the time until the reactor is scrammed, but then continues to rise quite slowly as a maximum value is reached and the heat generation continues to decrease. Finally, the film boiling condition is overcome resulting in a large decrease in surface temperature and an increase in flow rate to a condition of single phase cooling.

RESULTS

The results of the transient experiments simulating a front face fitting failure are presented in Figure 1 as 'Maximum Heater Rod Surface Temperature' versus 'Rear Header Pressure' for various initial tube power levels. Each of the data points on Figure 1 required an individual transient test of the type illustrated by Figure 3. A total of 44 such tests were made to obtain the data of Figure 1. 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.

Experimental Heat Input

The data of Figure 1 show two curves for each initial tube power. The difference between the two curves results from using different programmed power input relations during the simulated scram, as described above. A value of 5 per cent of the original equilibrium tube power has been estimated as the sensible heat release rate from the graphite stack under conditions of constant coolant flow rate during a scram. (3)(2) Under conditions of reduced flow rate and higher coolant temperatures, the sensible heat release rate would be reduced somewhat. Generating 'extra' heat in the heater rod is presently the only way of allowing for graphite sensible heat in the experimental apparatus, but this technique will result in higher-than-prototypical heater rod temperatures. Because of the effect of coolant temperature on graphite sensible heat release rate and the effect of generating 'extra' heat in the simulated fuel, it is judged that the 'nuclear heat plus 5 per cent' curves represent an upper limit on the rear header pressure requirements. Similarly, it is judged that the 'nuclear heat only' curves represent a lower limit on rear header pressure requirements, although a small amount of 'extra heat'(graphite nuclear heat generation) is being generated in the heater rod and would cause slightly higher-than-prototypical heater rod temperatures.

All data points on Figure 1 are for a 3 second delay between simulated piping failure and the beginning of power reduction. The effect of scram

[redacted]

[redacted]

delay time on maximum heater rod temperature was investigated briefly in reference (1). For the present studies, it is estimated that the maximum heater rod surface temperature should be reduced by 0.10 times the initial tube power if the scram delay time is reduced to 2 seconds after simulated failure.

Effect of Assumed VSR Strength

A comparison of the data from these experiments with the data from the K inlet piping failure experiments (1) shows that the VSR strength which is assumed for the power decay curve has a strong influence on the rear header pressure which is sufficient to prevent fuel jacket melting. This is illustrated by tabulated data of Table I.

TABLE I
COMPARISON OF ALLOWABLE INITIAL AND RESIDUAL HEAT INPUT RATES FOR BDF AND K REACTORS

Basis: The initial (time 0) heat input rates in the table at the corresponding rear header pressure will result in a 1000 F maximum heater rod surface temperature.

Rear Header Pressure psig	Elapsed Time of Power Decay, seconds	K-Reactor 500 ih Decay Curves: Nuclear Heat Plus 5% KW	K-Reactor 500 ih Decay Curves: Nuclear Heat Only KW	BDF-Reactors 1100 ih Decay Curves: Nuclear Heat Plus 5% KW	BDF-Reactors 1100 ih Decay Curves: Nuclear Heat Only KW
20	0*	--	900	805	1075
	4	--	238	170	173
	10	--	176	136	128
	40	--	85	93	70
30	0*	865	1025	960	1340
	4	272	271	203	216
	10	213	201	162	159
	40	125	96	110	87
40	0*	995	1140	1095	--
	4	312	301	231	--
	10	245	224	135	--
	40	143	107	126	--
50	0*	1105	1240	1210	--
	4	347	328	255	--
	10	272	244	204	--
	40	159	116	139	--
60	0*	1210	1340	1320	--
	4	380	354	279	--
	10	298	263	223	--
	40	174	126	152	--

*The heat input rate was the same for the previous 3 second period between piping rupture and beginning of power decay.

[redacted]

The best comparison is between the residual heat input rates for the 'K reactor nuclear heat only' case and the 'BDF reactor nuclear heat plus 5 per cent' case. Comparing these values for various elapsed times of power decay, one finds that the power input during the initial (and critical) portion of the transient is nearly equal for these two cases. In fact, the power input for the 'K nuclear heat only' case with 500 ih scram is generally slightly greater than for the 'BDF nuclear heat plus 5 per cent' case with an 1100 ih scram. Thus, for VSR strength values which would result in the same heat input, whether this be for nuclear heat only or nuclear heat plus graphite sensible heat, the rear header pressure requirements for K reactor should be equal or slightly lower than the pressure requirements for BDF reactors. This means that if an 1100 ih scram were used for K reactor, the pressure required to prevent fuel jacket melting would be about 10-15 psig lower than those reported in reference (1) for the 'nuclear heat plus 5 per cent' case. This is illustrated in Figure 2. For the 'nuclear heat only case' the required rear header pressure would be reduced even more than 15 psig for K reactor depending on the initial tube power. Where it was originally thought that the big difference between the K data and the BDF data was because of the geometry differences, it now is apparent that the residual heat input differences would be the major contributing factor.

Effect of Rear Header Water Temperature

The effect of variations in rear header water temperature was investigated briefly in reference (1) and found to be small. From this standpoint, the data of Figure 1 are not overly conservative in application to the reactor.

Heat Capacity of Experimental Heater Rod

The heat capacity of the heater rod used in these experiments was 0.54 Btu/°F-per foot of length while the heat capacity of 0-II-N I&E fuel pieces would be about 0.36 Btu/°F-per foot of length. The problem of the effect of heat storage capacity on fuel temperatures during such transient heat transfer experiments was investigated analytically and discussed in reference (1). It was concluded 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 when the ratio of heat capacities was about 0.54/0.36 = 1.5. Thus, if one uses 1220 F (aluminum melting point) as the maximum allowable fuel surface temperature and 300 F as the surface temperature before 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 fuel.

General Observations

For the combinations of rear header pressures and initial tube powers given in Figure 1, the flow rate from the rear header through the tube ranged from about 0.5 to 5.0 gpm during time interval from simulated piping failure to recovery from film boiling conditions. Superheated steam discharge was not

[redacted]

encountered in any of the experiments. Steam discharge temperatures were generally 270 F or less in the front face nozzle during the film boiling portion of each experiment.

Maximum heater rod temperatures always occurred in the region of highest heat flux - an 8 foot length in the center of the heater rod. The specific location of the hot spot varied but generally occurred toward the normal upstream end of this high heat flux section. The absence of hot spots outside of the high flux section would indicate the absence of severe heater rod 'cocking' (eccentricity), at locations outside the center section at least, and would also indicate the absence of flow channeling or stratification. The application of the subject data to reactors would not be conservative if severe fuel element eccentricity existed.

While the experimental apparatus employed a BDF outlet fitting assembly, the effect of these outlet fittings is considered to be quite small during the low reverse flow rates involved during the period of maxinum fuel temperatures. Hence, these data should be applicable to DR and H reactors as well as B, D, F reactors.

E. D. Waters
Engineer
Thermal Hydraulics Operation

M. R. Kreiter
Engineer
Thermal Hydraulics Operation
HANFORD LABORATORIES OPERATION

EDW:MRK:leb

BIBLIOGRAPHY

(1) HW-68214 - "Heat Transfer Experiments Simulating Failure of the Inlet Piping to a K Reactor Process Tube", E. D. Waters, D. E. Fitzsimmons, 1/20/61.

(2) HW-33870 - "Heat Generation and Total Heat Output From the Pile After Shutdown", S. S. Jones, 11/23/54.

(3) HW-68273 - "Heat Transfer Experiments Simulating Front Header Pressure Reductions to a K-Reactor Process Tube", E. D. Waters, D. E. Fitzsimmons, 2/20/61.

[redacted]

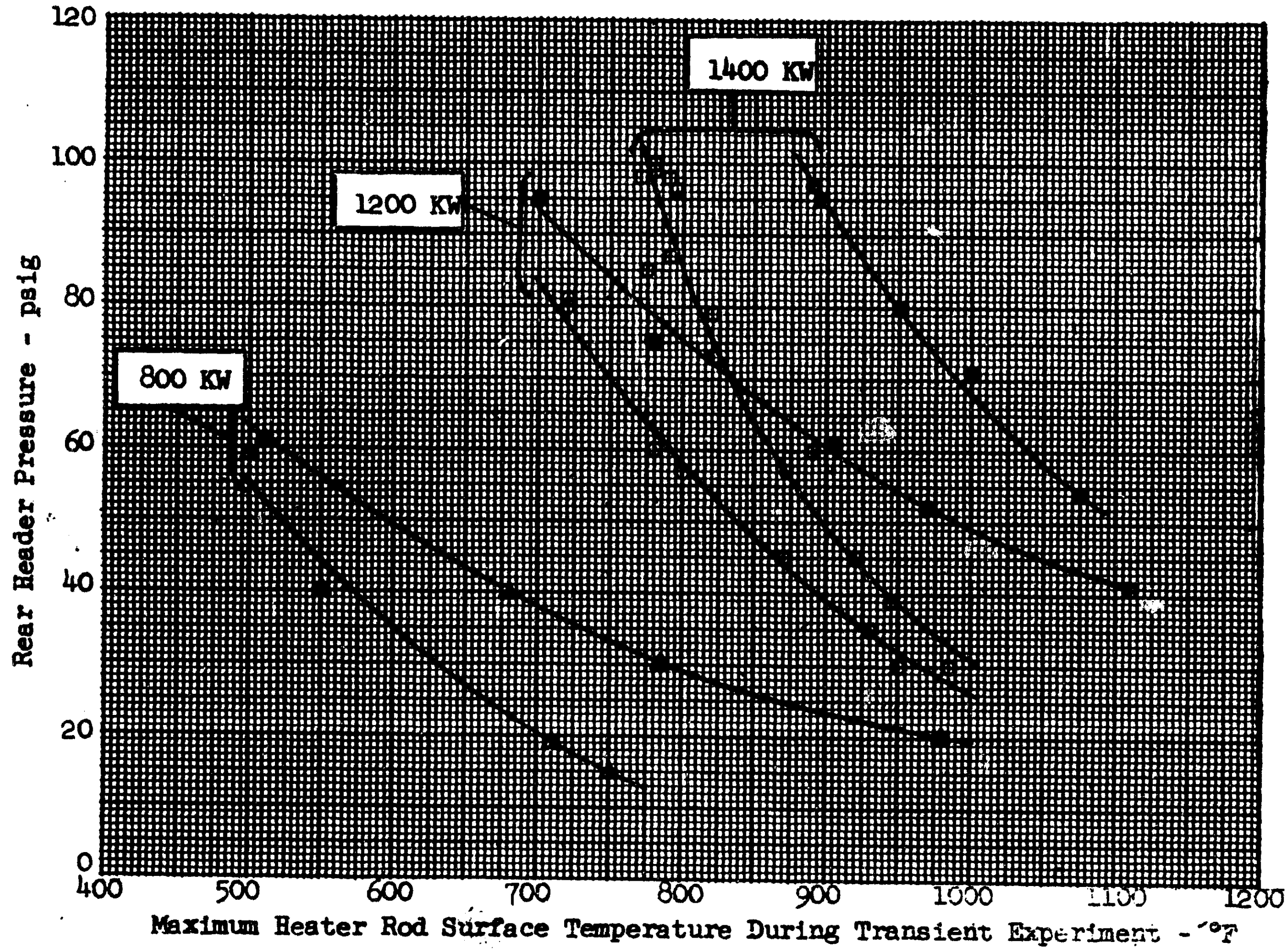
[redacted]

DECLASSIFIED

Figure 1

Results of Inlet Piping Failure Experiments

Basis: BDF-Reactor mockup with front pigtail open to atmosphere at header end. Temperature of water from rear header = 125 C for 1400 KW runs and 110 C for 800 KW and 1200 KW runs. Simulated 32 piece O-II-N fuel charge. See discussion regarding upper and lower curves for each initial tube power. 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 6.

[redacted]

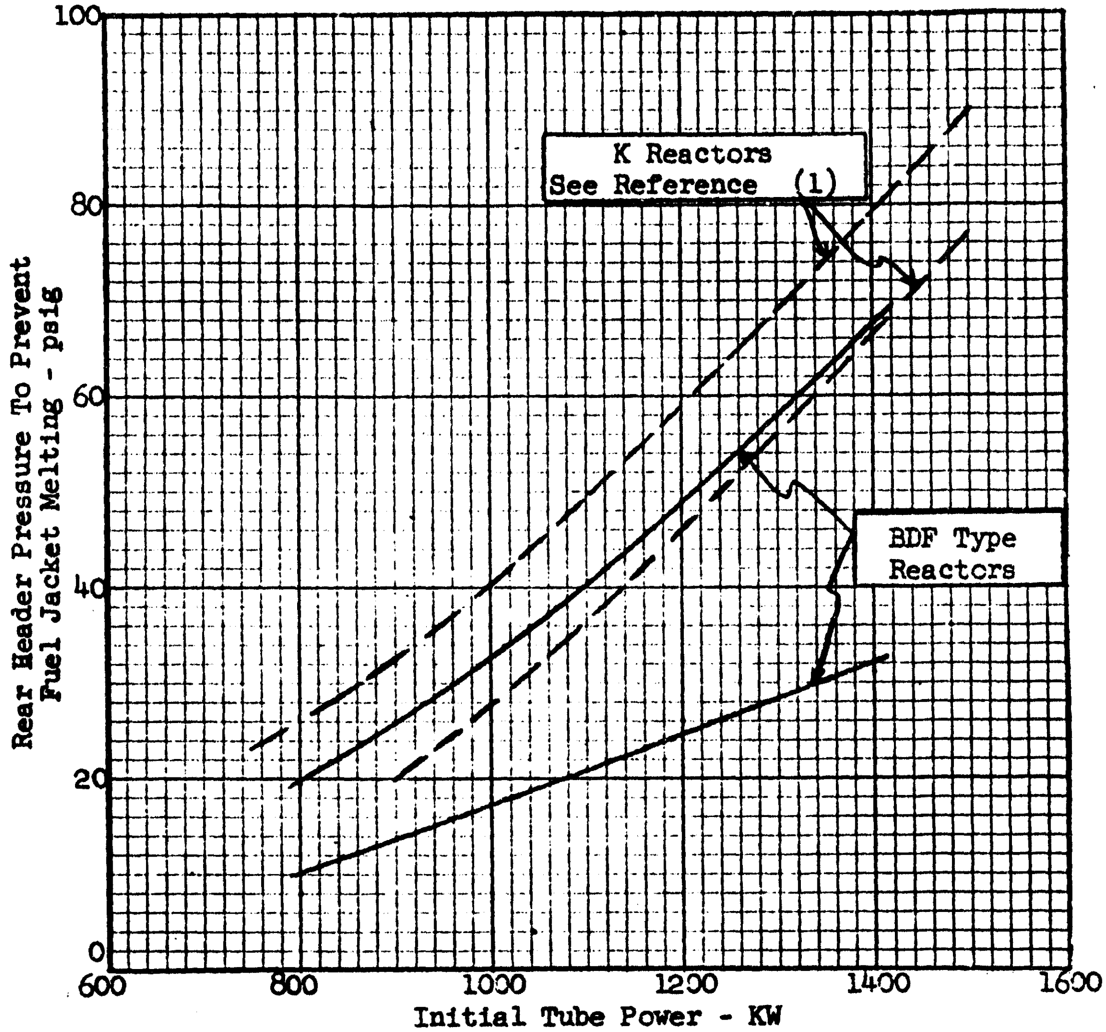
[redacted]

DECLASSIFIED

Figure 2

PRESSURE REQUIRED TO PREVENT FUEL JACKET MELTING AFTER AN INLET PIPING FAILURE

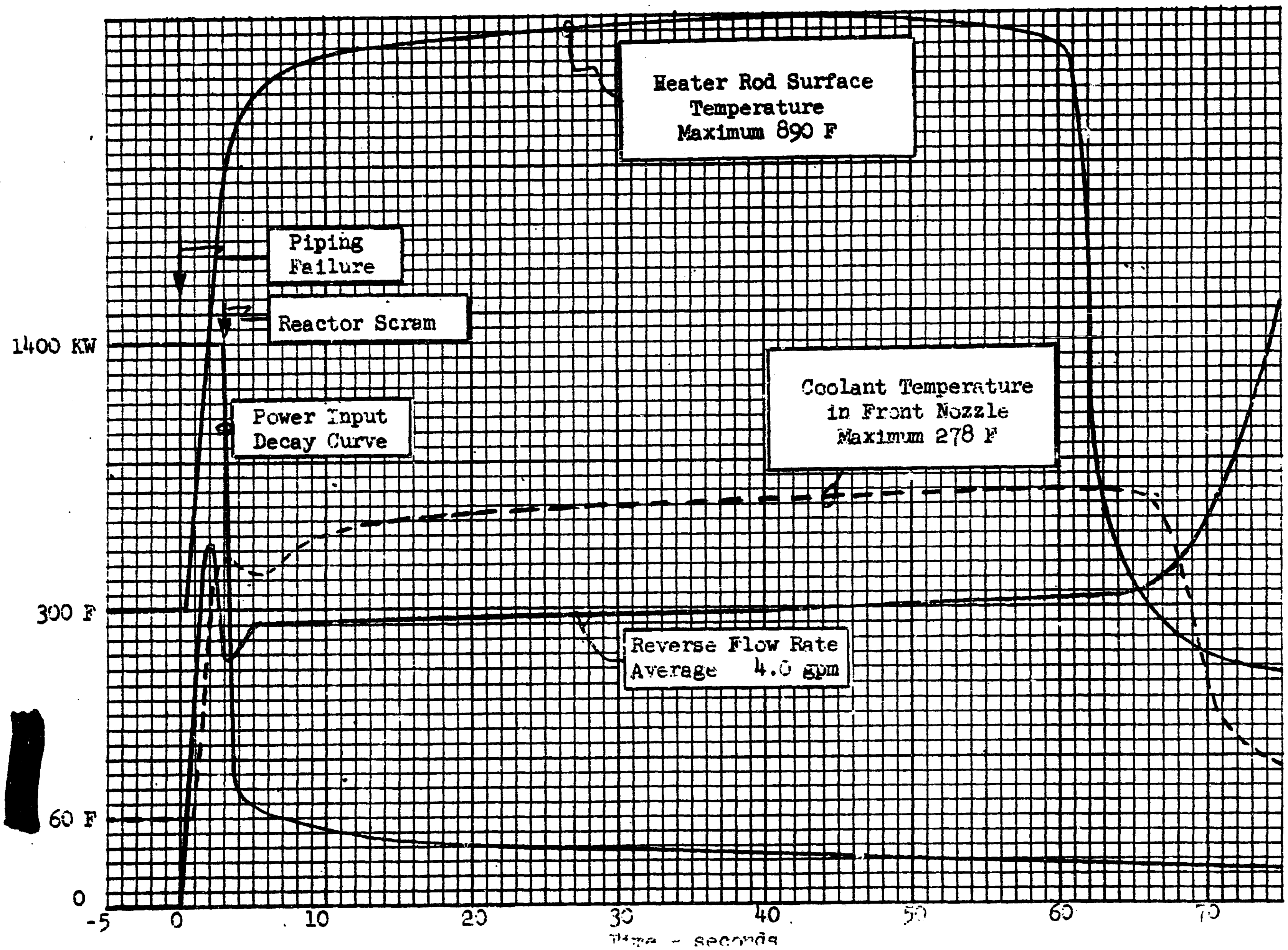
Basis: Same as Figure 1.



NOTE: The solid lines of this figure are cross-plots from Figure 1 at a heater rod temperature of 1000 F. This should be approximately equivalent to a 1220 F uranium fuel surface temperature.

[redacted]

[redacted]



[redacted]

Figure 3

SCHEMATIC REPRESENTATION OF DATA FROM A TYPICAL PIPING FAILURE TRANSIENT EXPERIMENT

Initial Tube Power = 1400 KW
Decay Curve = nuclear heat + 5%
Rear Header Pressure = 97 psig

DECLASSIFIED

DATE
FILMED
8/18/94

END