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HEAT TRANSFER EXPERIMENTS SIMULATING FRONT HEADER
PRESSURE REDUCTIONS TO A K-REACTOR PROCESS TUBE

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E. D. Waters and D. E. Fitzsimmons

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HEAT TRANSFER EXPERIMENTS SIMULATING FRONT HEADER PRESSURE REDUCTIONS TO A K-REACTOR PROCESS TUBE

E. D. Waters and D. E. Fitzsimmons

PURPOSE

This report presents the results of heat transfer experiments which were conducted to determine the ability of a K-reactor fuel channel assembly to undergo various degrees of coolant supply header pressure decrease without resulting in fuel jacket melting. The experiments were conducted with an electrically heated test section in the single tube prototype Heat Transfer Apparatus of Thermal Hydraulics Operation.

INTRODUCTION

An accidental reduction in reactor front header pressure will result in decreased flow of cooling water past the fuel elements. Although the reactor power level will be decreased almost immediately by insertion of the safety rods, excessive fuel temperatures can result in some cases. In particular, when the flow decrease is very severe or when boiling starts and a further self-induced flow reduction occurs, the rate of heat generation can be greater than the rate of heat removal by the available cooling water during the reactor shutdown transient. The objective of this study, then, was to determine the changes in fuel elements temperature resulting from various reductions of flow as caused by a sudden decrease in front header pressure. This was done by simulating reactor conditions in laboratory heat transfer experiments using electrically heated test sections.

Heat transfer experiments have been performed previously. (1, 2, 3)* investigating the effects of flow reductions on fuel temperatures. In these previous experiments the front header pressure was either held constant or cut immediately to zero and the results are applicable mainly to incidents that happen to a single process tube. (For example, plugging of the flow channel or rupture of an inlet flow line.) The experiments reported in this report were all run with various reductions in front header pressure as would result, for example from the loss of electrical power to the pumps or rupture of a front riser. As such, these events would affect all of the process tubes and fuel elements in the entire reactor.

An interesting problem arising in the laboratory experiments was the selection of a suitable heat generation rate decay curve to simulate the decrease in heat output following a reactor 'scram'. The heat output from the experimental apparatus was measured as a function of time after 'scram' for a constant flow rate condition and for a constant outlet temperature condition. The results are presented as an appendix to this report. Where possible, these results were compared to similar reactor measurements.

SUMMARY

Events concerned with flow reductions as caused by a sudden reduction in front header pressure or by a gradual reduction in front header pressure were

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simulated on a single tube prototype K reactor process assembly in the Heat Transfer Apparatus of Thermal Hydraulics Operation. The resultant data are presented in this report along with a description of the apparatus and the bases and procedures which were used in the experiments. These data should be useful in evaluation of the reactor hazards which involve the coolant supply systems.

The results of the sudden pressure reduction experiments are presented on Figures 1 and 2. It is not the purpose of this report to make recommendations regarding the coolant supply systems. However, as an example of the use of the experimental data, the following sample conclusions can be reached: Comparison of these data with the postulated sudden pressure loss conditions of Table I, page 5 indicates that; (1) with six pump operation at a K reactor, fuel jacket melting should not result even with failure of a front riser at tube powers up to about 1900 KW, and (2) with five pump operation the situation is more critical and a sudden pressure reduction event as serious as the rupture of a supply line to a front riser could result in fuel jacket melting at tube powers of about 1500 KW.

The gradual pressure loss experiments simulated a complete loss of electrical power to one K reactor with loss of electrical power to the primary pumps of the other K reactor. The study indicated that the beginning of boiling would initiate flow instability and result in operation with film boiling conditions on the fuel surface. Although the experimental simulation is relatively poor at 80 to 100 seconds more after scram because the graphite sensible heat release is simulated by generating extra heat in the heater rod, the results do indicate that to count on cooling-by-boiling (or more accurately cooling-by-steam) at cross-tie emergency cooling system supply pressures would risk melting of fuel jackets.

The Appendix of this report concerns the heat output from the experimental equipment in simulation of a reactor scram and power decay. The data for a simulated scram on the Heat Transfer Apparatus is compared with a recommended curve of reactor heat output which is based on a number of reactor experiments. The comparison shows good agreement when some extra heat is generated in the heater rod to compensate for sensible heat release from the graphite stack associated with a reactor lattice.

NOTE CONCERNING APPLICATION OF THE LABORATORY EXPERIMENTS TO REACTOR CONDITIONS

A question arises as to the validity of using results from laboratory experiments using a single process tube to predict events in a reactor that has thousands of tubes. The concern here is that with multiple parallel flow channels as exist in a reactor, the cooling water will seek those tubes with the lowest flow resistance and "starve" those tubes in which boiling takes place. While this concern is valid, it should be realized that a very accurate prediction as to conditions in an entire reactor following an event as considered here is very difficult. This problem has been discussed in general terms before (4) but not in detail since the results are so very dependent upon a number of initial conditions of the reactor, as well as such

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unknown conditions as possible parallel channel pressure - flow pulsations and static pressure changes caused by bulk boiling in the outlet piping.

In any case, it is thought that the conclusions presented here in regard to the application of the experiments to reactor conditions are fairly valid. The header pressures used in the sample conclusions and reported in Table I on page 5 are based on the pump supply curves at normal reactor flow rates. If boiling were to take place in the reactor, the front header pressures should be at least as great as those presented in Table I and, if the boiling caused an appreciable decrease in total reactor flow, then the front header pressure should be greater than those of Table I. While some tubes might be worse off than others, no tube on the reactor should experience conditions any worse than those run in the experiments.

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 simulated a 38 piece charge of K-III I&E fuel pieces. For a detailed description of the apparatus see page 4 of reference (3).

The operating procedure consisted of first establishing equilibrium flow rate and coolant outlet temperature at the selected tube power level. Then the front header pressure was suddenly reduced to a pre-selected value and controlled at that condition through the remainder of the run. Three seconds after header pressure reduction, the power input to the test section was reduced systematically to simulate a 500 ih reactor scram (as discussed in the Appendix).

Heater rod temperatures were monitored to determine whether adequate 'fuel' cooling would be achieved during the post-scram period with the new reduced header pressure. In general, if heater rod temperatures exceeded ~ 550 C, the power input was manually reduced to zero and the run was labeled as 'no recovery' (of adequate cooling). Occasionally rod temperatures in excess of 550 C were allowed without manual power reduction if the rate of temperature rise was low as the temperature approached 550 C.

RESULTS

Experimental data from tests simulating a sudden reduction in front header pressure are presented on Figures 1 and 2. Figure 1 compares maximum heater rod surface temperature versus the reduced header-to-header pressure drop for various initial tube powers. Figure 2 is essentially a cross-plot of selected values from Figure 1 to show the minimum header-to-header pressure drop which would prevent uranium fuel jacket melting for various initial equilibrium tube power conditions. This cross-plot includes an adjustment of temperatures to compensate for the difference in heat storage capacity between the experimental heater rod and actual uranium fuel pieces (3).

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The results of tests simulating a gradual reduction in front header pressure to cross-tie backup pressures were not amenable to graphical presentation, but rather are discussed on page 7.

DISCUSSION

The problem examined in this study is the definition of the allowable decrease in reactor coolant supply header pressure without resultant reactor damage by excessive fuel jacket temperatures. The possible reactor events which could lead to such a supply pressure decrease include:

- (1) Loss of a pump shaft or rupture of a pump discharge line.
- (2) Rupture of a main coolant line (riser).
- (3) Electrical power failure to process pump motors.

Events (1) and (2) should result in a sudden decrease in coolant supply pressure to the reactor, whereas event (3) should result in a gradual decrease in pressure as the inertial energy of the pump flywheels is used. Laboratory experiments were performed to simulate these two types of pressure decrease. While such events would affect many or all of the parallel flow fuel coolant channels on a reactor, the experiments were limited to the study of a single coolant channel while undergoing such supply pressure transients.

Sudden Loss of Front Header Pressure

The following values have been estimated by Research and Engineering personnel for various sudden supply pressure decrease events (6, 11):

TABLE I

No. of Primary Pumps Operating	Event to Cause Pressure Decrease	Front Header Pressure, psig		Rear Header Pressure, psig (after event)	Available Header-to-header ΔP , psi (after event)
		Before	After		
6 5	Loss of one pump shaft	355 280	262 209	38 32	300 177
6 5	Loss of two pump shafts	355 280	210 113	32 27	178 86
6 5	Rupture of riser supply line	355 280	227 80	35 27	192 53
6 5	Rupture of riser proper	355 280	157 57	30 25	127 32

Thus with these postulated equipment failure events, header-to-header pressure drops as low as 32 psi might be encountered.

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Sudden header pressure loss events to various header-to-header pressure drop values were then simulated with the electrical power input beginning to decrease in accordance with a 500 ih vertical safety rod (VSR) scram at 3 seconds after pressure loss. Two tube power decay curves were used for each power level; one decay curve allowed a heat input which equals the nuclear heat generation rate (nuclear only) while the other decay curve allowed a heat input which also provides for sensible heat removal from the graphite stack during the power decay period (nuclear + 5%). (These two decay curves are discussed in the Appendix, page 12. The appendix also includes a comparison of data for heat output from the experimental apparatus with the recommended heat output curve from the reactors.)

Although electrical generation of extra heat to compensate for graphite sensible heat release is not entirely satisfactory, the two power decay curves should encompass the upper and lower limit of heat transfer to the coolant which may occur in a reactor channel for a 500 ih scram. If 500 ih is too low an estimate of VSR strength, the 'nuclear only' curves of Figures 1 and 2 should correspond fairly well to an 800 ih scram with an allowance for graphite sensible heat over the time interval which is of interest in sudden pressure loss events.

The results of the sudden pressure loss tests are presented in Figure 1. Before the data of Figure 1 are applied to reactor hazard analysis, the maximum heater rod surface temperatures should be adjusted to reflect the difference in heat storage capacity between the experimental heater rod and reactor uranium fuel pieces. This adjustment is discussed in reference (3). In this reference it was concluded that a heater rod surface temperature of 960 F to 1010 F would correspond to a uranium surface temperature of 1220 F (aluminum melting). Applying this adjustment, information on Figure 1 is then cross plotted on Figure 2 to show the header-to-header pressure drop which would be required to prevent uranium fuel jacket temperatures from reaching 1220 F (i.e. 960 F heater rod surface temperatures) under conditions of sudden supply header pressure reductions.

The ΔP values of Figures 1 and 2 are based on experiments at a rear header pressure of 20 psig but should not be appreciably different for rear header pressures in the range 20 to 60 psig. One test run which was conducted at 45 psig rear header pressure resulted in only a slight change in heater rod surface temperature. Initial outlet water temperatures which are lower than those used in the experiments (125 C) should result in slightly lower maximum heater rod surface temperatures. This was confirmed by one test run with an initial outlet temperature of 100 C.

The data of Figure 1 are of the same type as those presented in reference (3) for inlet piping failure incidents; front header pressure reductions result in lower heater rod surface temperatures, however. This is reasonable for two reasons. First, there is not a flow stagnation and flow reversal for the front header pressure reduction. This eliminates a time lag and also

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assures that cooler water would be reaching any particular heater rod location. Second, the back pressure for the front header pressure reductions was higher which would result in smaller vapor volumes, and thus allow a higher flow rate for any given inlet-to-outlet pressure drop,

Comparing values on Figure 2 with values from Table I and recalling that these are single tube experiments, it appears that there should be no problem in preventing fuel jacket melting with 6 pump operation at tube powers up to at least 1900 KW. But for 5 pump operation, the situation is more critical and any sudden pressure reduction event as severe as the rupture of a supply line to a front riser could result in fuel jacket melting at tube powers of about 1500 KW.

It is worthy to note the relation between the hydraulic supply and demand curves during a sudden header pressure reduction. This is illustrated in Figure 3 which includes an approximate representation of generalized boiling curves presented in reference (5). Note that the highest 'reduced' front header pressure of the present experiments would result in a supply curve which would be appreciably below the minimum pressure on the boiling demand curves. Thus, in all of the sudden pressure loss cases a condition of flow instability was well established before the power input reduction was started. But a complete recovery of adequate cooling was attained without excessive heater rod temperatures. For the 1000 KW (nuclear only) case, a complete recovery was attained with a supply header pressure as low as 35 psig (curve A of Figure 3). Thus, as was stated in reference (6), the occurrence of flow instability does not necessarily mean fuel melting. And operation in a manner that would prevent reaching the condition of unstable flow during a sudden reduction of front header pressure would result in conservative operating limits.

Gradual Loss of Front Header Pressure

The process coolant supply system of a Hanford production reactor consists of three independent means of providing reactor cooling. [A description of the K reactor systems is included in reference (7)]. At a K reactor there exists the remote instance that failure of the primary electrical supply to the process pumps of both K reactors would be followed by failure of the back-up steam turbo-generator system to supply electric power to the low-lift process pumps of one 'severely stricken' reactor. In this instance, the tertiary or last-ditch coolant system should supply coolant at very low pressures from the other K area reactor through a coolant cross-tie pipeline between the K reactor pair.

The coolant supply pressure and flow to such a stricken reactor would gradually decrease over a period of many seconds (about 140) and then would hold essentially constant for an extended period or until the primary electrical supply was restored. During the first few seconds following such an incident, a rise in reactor coolant outlet temperature occurs because the heat output decrease does not begin until 2 or 3 seconds after the flow

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decrease begins. This initial temperature transient has been discussed elsewhere (6). In fact the sudden pressure reduction experiments described above can be used to minimize the significance of the short period of 'inadequate' cooling which occurs at the beginning of a primary electrical power outage. Hence, the discussion from hereon will be concerned with the transient conditions near the time where the equilibrium or minimum flow rate occurs--about 100 to 150 seconds after scram.

The question concerned with this period of time is 'Can the reactor be cooled-by-boiling during this crucial period or must the emergency coolant system be adequate to maintain sub-saturated coolant outlet conditions'. Without further suspense, it is concluded from the experimental studies that cooling-by-boiling is not feasible on a single tube basis during the postulated gradual pressure loss event.

Gradual supply pressure decrease events were simulated in the laboratory. The heat input was programmed for a time interval of 750 seconds in accordance with a 500 ih scram plus 5 per cent of original tube power for graphite sensible heat release. The supply pressure decrease was programmed in accordance with data of Figure 23 (corrected) of reference (8). Assuming negligible pressure drop from riser to the header, there should be about 6 psi header-to-header pressure drop available at equilibrium or minimum flow conditions after the gradual pressure decrease. This gives an equilibrium tube flow of about 5 gpm (cold). Using this ΔP as the equilibrium value, test runs were made as indicated in Table II. The occurrence of boiling was detected during any run by small rapid oscillations in heater rod surface temperatures.

TABLE II

Basis: Equilibrium ΔP (header-to-header) = 6 psi
Heat input = 500 ih scram (nuclear + 5% - See Appendix)

Run No.	Initial Conditions			Rear Header Pressure, psig	Equilibrium flow rate with no heat input, gpm	Did boiling occur?	Was recovery made without dropping final power?
	Front Hdr.Press, psig	Tube Power, KW	Outlet Temperature °C				
VII-b	350	1000	125(qual.)	15	2.3	yes	no
VII-c	350	1000	80	15	5.1	no	yes
VII-d	350	1500	110	15	5.1	no	yes
VII-e	400	2000	130(qual.)	15	5.1	yes	no
VII-f	400	2000	130	50	5.1	no	yes

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The qualitative results in Table II show that when boiling occurred, recovery to low heater rod temperatures was not attained without dropping the power input below the programmed value. Continued operation at the programmed power input would have resulted in heater rod damage.

This situation can be illustrated by sketches, as on pages 3, 4, and 5 of reference (4), or can be described briefly as follows: Once the supply pressure curve drops below the minimum inflection point (which corresponds to beginning of boiling) on the demand curve, then the flow rate begins to decrease toward a very low value which would correspond to film boiling conditions with very high heater rod ('fuel') temperatures and probably even superheated steam discharge. The time required to reach high 'fuel' temperatures depends a great deal on the heat generation rate. But once boiling begins the only way to 'recover' to low fuel temperatures is for the heat generation rate to decrease until the peak of the demand pressure curve falls below the supply pressure curve; that is, recovery cannot begin until the constant equilibrium supply pressure becomes an excess header pressure. But at about 140 seconds after scram, the heat generation rate does not decrease fast enough to permit recovery before excessive 'fuel' temperatures are attained.

This situation is verified by data on Figure 3 of reference (9). In the reference experiments, flow was stopped to a process coolant channel until various periods of time after boiling had begun. At a tube power of 15 KW, boiling was allowed for about 200 seconds and recovery was attained with only 3.6 psi ΔP . But with a tube power of 40 KW, boiling could only be allowed for 3 to 4 seconds before re-application of pressure if recovery was to be attained with 3.6 psi ΔP . Since the residual heat release rate at 140 seconds is about 95 KW for a 1000 KW tube power and 190 KW for a 2000 KW tube power, it is indicated that boiling could not be allowed to take place with the comparably small pressure drop which is available with the K reactor last ditch coolant supply system.

A similar type of test run was made during the present study. A stable bulk boiling condition was set up in the prototype assembly by temporarily reducing the supply pressure and then re-applying a pressure drop of 6.8 psi across the assembly while the tube power was held constant at 50 KW. (The rear header pressure was 0 psig.) The outlet fluid temperature was then at saturation conditions and the heater rod surface temperatures were near saturation temperatures indicating that the system was operating at reduced flow with stable bulk boiling.

The tube power was then increased to 55 KW and the rod surface temperatures increased slowly into film boiling conditions. A steady state condition was not reached. When the surface temperatures reached approximately 1150 F (about 8 minutes) and the tube outlet steam was superheated above 600 F, the power was reduced to 50 KW. This caused the temperatures to level off. But after operating under these conditions for one half hour, the surface

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temperatures had decreased only about 50 F and there was no real indication of returning to the original surface temperatures near water saturation temperature. The power was then dropped to 45 KW and the system completely recovered to non-boiling conditions in about 10 minutes.

Although there probably exists some very low tube power wherein the heat could be removed by cooling-by-steam without excessive fuel temperatures, this experiment described above would indicate that this tube power would be less than 50 KW for a 6.8 psi pressure drop to atmosphere. It must be pointed out that, while the heat source for the experimental equipment is the heater rod, in the reactor the graphite is the major heat source at low tube powers. Thus the surface temperatures measured in the experiments would not apply to fuel element surface temperatures for times after scram of 80 to 100 seconds or more when the majority of the heat is from the graphite.

From the gradual pressure loss type experiments, it is concluded that initiation of boiling in a process tube, by simultaneous power decay and flow decrease, will result in flow instability at the low equilibrium supply pressures postulated for this event. While the experimental simulation may cause unrealistically high heater rod surface temperatures, the undesirability of allowing boiling during a gradual pressure loss is indicated. Since it seems unwise to allow boiling on a single tube basis, it certainly seems unfeasible to allow boiling on a whole reactor basis where the unknown effects of parallel channels and boiling in the rear risers would be introduced, at least until more is known of these effects.

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BIBLIOGRAPHY

- (1) HW-56621 - "Experimental Results of Tests Simulating Plugging of a K Tube With I&E Slugs," D. E. Fitzsimmons, G. M. Hesson, July 7, 1958.
- (2) HW-60518 - "K Reactor Panellit Protection Against Plugging Incidents With Low Front Header Pressure," D. E. Fitzsimmons, G. M. Hesson, June 1, 1959.
- (3) HW-68214 - "Heat Transfer Experiments Simulating a Failure of The Inlet Piping to a K Reactor Process Tube," E. D. Waters, D. E. Fitzsimmons, January 20, 1961.
- (4) HW-64368 - "Some Comments on Thermal Hydraulic Conditions at K Reactors Following Emergency Shutdowns," G. M. Hesson, March 16, 1960.
- (5) HW-64029 - "Extended Hydraulic Demand Curves for K Geometry Tubes With I&E Fuel Elements," G. M. Hesson, D. E. Fitzsimmons and M. F. Kenninen, February 25, 1960.
- (6) HW-61760 - "Potential for Process Tube Burnout During Transient Conditions," P. A. Carlson, S. S. Jones, April 22, 1960.
- (7) HW-64417 - "Review of the Status of Hanford Reactor Emergency Cooling," S. S. Jones, June 1, 1960.
- (8) HW-59892 - "K Reactor Hydraulic Working Curves and Numerical Results of PT-IP-130A K Reactor Trip-out Test Down to Cross-Tie Back-up," J. L. Benson, November 18, 1959.
- (9) HW-66274 - "Pressure Required to Overcome Boiling at Low Tube Powers - BDF Reactors," E. D. Waters, D. E. Fitzsimmons, August 5, 1960.
- (10) HW-33870 - "Heat Generation and Total Heat Output From the Pile After Shutdown," S. S. Jones, November 23, 1954.
- (11) Personal communication with S. S. Jones, May 1960.

APPENDIXHEAT OUTPUT FROM THE HEAT TRANSFER APPARATUS
AND COMPARISON WITH RECOMMENDED REACTOR HEAT OUTPUT CURVESUMMARY

Experiments were conducted to determine the heat output from the experimental Heat Transfer Apparatus during a simulated reactor scram. For these experiments, the heat input to the simulated fuel (heater rod) was selected to approximate the sensible heat release as well as the heat generated by nuclear processes after scram. The results are compared to a curve of reactor heat output which has been recommended for use in analysis of the present Hanford emergency coolant systems.

INTRODUCTION

The nuclear heating which occurs in the fuel pieces of a reactor is simulated in the Thermal Hydraulics Operation heat transfer apparatus by direct current resistance heating in "heater rods" (or tubes) which are geometrically similar to the nuclear fuel pieces. During steady state heat transfer experiments, the heat generation rate in the heater rod is selected and adjusted by manual movement of the power supply controls. During transient heat transfer experiments, pre-selected heat generation rate variations can be accomplished by means of a Moseley Autograf X-Y Recorder (Model 2S) with a curve follower attachment. The Moseley instrument essentially controls the d.c. power supply according to a pre-drawn curve of power input versus time. Then the problem arises of determining a suitable curve of power input versus time to simulate conditions after a reactor scram.

The rate of heat removal from a reactor by the coolant following a scram is influenced by a number of variables. The heat which is removed might be divided as generated heat and sensible heat. The amount of heat generated after a scram is affected by numerous nuclear physics considerations. The sensible heat content of the system is essentially a function of the relative reactor components temperature and the water temperature. Thus, the heat available for transfer to the coolant from these two sources should lend itself fairly well to analytical determination. The difficulty in predicting the rate of heat output then arises from the fact that the heat transfer coefficients, the water temperature, and the temperature-difference driving force are continuously changing with time after the scram. Even when using certain experimentally determined empirical constants, it has been shown that there can be considerable difference between the predicted total heat removal rate from a reactor lattice and the measured heat removal rate during a typical reactor scram. (10)

The Low Pressure Heat Transfer Apparatus is a fairly accurate geometric prototype of a single reactor fuel and coolant channel from tube inlet to the rear header. As such, the assembly has a sensible heat capacity

associated with each component just as the actual reactor channels, with two exceptions; first, there is no graphite stack or radiation shield around the laboratory apparatus to provide sensible heat storage; second, the heat storage capacity of the electrical "heater rods" which are used in the experiments is greater than for the fuel column of a reactor process channel. The provision of a simulated graphite stack on the laboratory apparatus seems prohibitive from economic and mechanical standpoints. Since some allowance must be made for this condition, a recourse is to generate extra heat in the "heater rod" to make up for the heat which might be transferred from the graphite to the coolant in a reactor channel. This is not entirely satisfactory because; (1) the sensible heat transfer in the reactor would be affected to a degree by coolant temperature changes during a flow transient, while electrical heating is not, (2) at present, there are no completely satisfactory data to show the rate of sensible heat release from the graphite stack, and (3) generating extra heat in the heater rod causes heater rod temperatures to rise faster and to higher values than would result if this heat were passing from the graphite through the process tube and into the coolant during flow transients. Some relief from the latter condition is provided by the fact that the heat storage capacity of the heater rod is greater than for the reactor fuel elements. This condition is discussed in the appendix of reference (3).

Selected Basis for Heat Input

The actual amount of heat to be generated in the heater rod during the heat output experiments was selected to be the total theoretical nuclear heat generated in fuel and graphite (assuming a 500 ih scram) plus 5 per cent of the original tube power before scram. Thus, 5 per cent of the original tube power represents the allowance for sensible heat release from the graphite. Curves of power input versus time were constructed on this basis and were used to control experimental heat input during simulated reactor scrams.

This selection was made on the basis of information in HW-33870 as follows: At 100 seconds after a 500 ih scram, the total theoretical heat generation equals about 5.0 per cent of the original power level as shown on Figure 1. Table IV, page 23, indicates that only 0.20 of this would be generated in the graphite and the remainder would be generated in the fuel. Therefore, only about 1 per cent of the original power is generated in the graphite at 100 seconds after scram. Total heat output from the graphite is about 6 per cent of original power as determined by graphite (temperature) period* measurements made on the reactors and shown in Figure 3. Thus, the difference between total heat released from the graphite and the heat generated

*The graphite temperature period is the time required for the graphite temperature to decrease by a factor of $e = 2.718$, so that the graphite temperature at any time might be represented by

$$T_{\theta} = T_1 e^{-\theta/T_g}$$

where T_1 is initial graphite temperature, T_{θ} is temperature at time θ , and T_g is the graphite period.

in the graphite (in this case 5 per cent) is assumed to be the sensible heat release. At other time intervals in the range 10 to 1000 seconds after scram, one finds approximately the same 5 per cent of original tube power as the difference between total heat output from the graphite and heat generation rate in the graphite. This appears to be the best information available for estimating the sensible heat release rate from the graphite stack.

Results of Heat Output Experiments

Using the heat input relations described above, experiments were conducted under two different procedures during which the heat output to the coolant was measured as a function of time after a simulated reactor scram. The results of these tests were compared with heat output data from reactor experiments and with the heat input curves which were used.

The first and simplest experiment was conducted by maintaining a constant front header pressure (nearly constant flow rate) and monitoring the flow rate and the water temperature rise through the apparatus as a function of time after simulated scram from some equilibrium operating tube power. The results of this test are presented on Figure 4 as curve D. It should be compared with curve E, which is based upon a number of experimental tests with essentially constant flow rate through the reactor following a scram. This curve has been recommended for use in analysis of Hanford reactor emergency coolant systems (7). It is pleasing to note that these curves compare quite well.

Next, experiments were carried out such that the outlet water temperature was maintained fairly constant by reducing coolant flow during the course of the 'scram' and power decay. This procedure would eliminate the majority of the sensible heat release to the water from the experimental apparatus components and the heat output to the coolant should approach the programmed heat input. (There will still be some sensible heat release as the radial and axial temperature gradients in the heater rod decreases with a decrease in heat generation rate.) The results of this test are shown as curve C of Figure 4 while the programmed power input is curve A. These curves also show good agreement. The divergence near the conclusion of the run probably represents the limit of accuracy in the read-out of the recorder chart based on a percentage of the original tube power. Curve B of Figure 4 is a plot of actual power input as indicated by the electrical power recorder. The oscillatory nature of this curve is due to the dynamic response characteristics of the controller which regulates the a.c. power supplied to the rectifier system. (Subsequent control circuit modifications have appreciably reduced these oscillations.) Although the deviations from the desired power input appear to be quite large in the first few seconds after scram, the time scale is greatly magnified here and the integrated energy input does not deviate appreciably from the desired value.

It was concluded from these experiments that (1) there is good agreement between the programmed power input and the actual power input to the

experimental apparatus, and (2) a programmed input equal to total theoretical nuclear heat generation plus 5 per cent of original tube power results in a curve of heat output which closely approximates that from reactor experiments under similar conditions of nearly constant flow rate. These experiments do not indicate that 5 per cent of the original tube power is the exact amount of sensible heat transferred from the graphite stack during a reactor scram nor do the experiments give any idea of how the sensible heat transfer from the graphite might vary with coolant temperature. Actual reactor experiments to better define these effects are currently being planned (11).

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FIGURE 1 - MAXIMUM HEATER ROD SURFACE TEMPERATURE DURING TRANSIENT EXPERIMENTS

Bases: 3 seconds delay time between pressure loss and 'scram'

Rear header pressure - 20 psig or more

Initial outlet water temperature - 125 C

Parameters - initial tube powers - See DISCUSSION, page 6, regarding two curves for each initial tube power.

Symbols: RNA - indicates recovery of adequate cooling was not attained without reduction of power input below programmed value.

NOTE: Uranium fuel surface temperatures will be higher than heater rod surface temperatures; see page 6.

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FIGURE 2 - PRESSURE REQUIRED TO RECOVER ADEQUATE COOLING DURING SUDDEN FRONT HEADER PRESSURE REDUCTION

NOTE: This Figure is a Cross-Plot From Figure 1 at a Heater Rod Temperature of 950 F. This Should Be Approximately Equivalent To a 1200 F Uranium Fuel Surface Temperature (See Page 6).

BASES: Same as For Figure 1.

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FIGURE 3 - RANGE OF EXPERIMENTS WITH REDUCED SUPPLY PRESSURE CURVES

Curves D, E, F are only approximate representations of steady state hydraulic demand curves given in reference (5).

FIGURE 4 - COMPARISON OF EXPERIMENTAL HEAT OUTPUT AFTER SCRAM

- Curve A - Programmed Heat Input (equal to total theoretical nuclear heat generated with 500 ih scram plus 5 per cent of initial tube power).
- Curve B - Actual Experimental Heat Input From Power Recorder
- Curve C - Experimental Apparatus Heat Output With Coolant Flow Throttled to Maintain Constant Outlet Temperature.
- Curve D - Experiment Apparatus Heat Output With Coolant Flow Rate Held Essentially Constant At Initial Value.
- Curve E - Recommended Heat Output From Hanford Reactors After VSR Scram (7).

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