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TECHNICAL NOTE ON THE DETERMINATION OF
K-REACTOR LAST-DITCH COOLING ADEQUACY

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TECHNICAL NOTE ON THE DETERMINATION OF
K-REACTOR LAST-DITCH COOLING ADEQUACY

By

Stephen S. Jones

February 28, 1964

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N O M E N C L A T U R E

- $\overline{F}_C$ - Fraction of clean line flow to stricken reactor because of corrosion.
- F_E - Flattening efficiency, ratio of temperature rise in ten hottest standard tubes to the bulk effluent temperature rise.
- 1 + F_G - Low flow error factor, fractional increase in measured flow to allow for flow measurement errors.
- F_S - Fraction of measured flow that can be counted on during an emergency because of possible crossheader strainer plugging.
- F_T - A flow temperature correction factor.
- f_G - Fraction of full level flow at τ seconds after a loss of pumping power.
- f_τ - Fraction of original heat output, or heat extraction requirements at τ seconds after a loss of pumping power and reactor scram.
- G_A - Assured flow to stricken reactor - 1000 gpm.
- G_M - Measured flow to stricken reactor - 1000 gpm.
- G'_M - Measured flow to aiding reactor (other K Reactor) - 1000 gpm.
- G_O - Original full-level reactor flow - 1000 gpm.
- P_B - Reactor power level with adequate bulk effluent cooling after a BPA loss down to last-ditch backup - Mw.
- P_O - A reactor power level for which adequate last-ditch flow should be determined - Mw.
- P_t - Reactor power level with adequate process tube cooling after a BPA loss down to last-ditch backup - Mw.
- T_i - Process water inlet temperature - $^{\circ}\text{C}$.
- T_{ic} - Crosstie water inlet temperature - $^{\circ}\text{C}$.
- X - Years since line cleaning.
- α - Lumped pumping flywheel decay parameter .
- τ - Time after reactor safety rod scram - seconds.

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February 28, 1964

TECHNICAL NOTE ON THE DETERMINATION OF
K-REACTOR LAST-DITCH COOLING ADEQUACY

By: Stephen S. Jones

INTRODUCTION

The Hanford reactors contain a very large inventory of fission products and it is very important that this inventory be contained. In recognition of this, rather stringent reactor cooling requirements have been established for the Hanford reactors.^(1,2) This includes the requirement of three independent cooling systems, including the system for full level operation. The second and last-ditch systems both are usable only after a reactor scram and take-over automatically from the flywheel decay of the process pumps.

In 1962, two projects were completed that affected the last-ditch cooling adequacy of the K reactors. One project increased the reactor pumping capacity and, therefore, its power level; and the other considerably modified the emergency cooling systems. Since this time, three tests have been conducted to determine this last-ditch cooling adequacy. In order to provide a uniform approach to this adequacy determination, the method employed by Reactor Engineering is presented herein.

SUMMARY

The report presents a method for determining the maximum reactor power level for which measured last-ditch flows will provide adequate emergency cooling. A number of pertinent charts are included. A method is also presented for determining the last-ditch flow that should be measured during a test for any desired level of normal reactor operation.

DISCUSSION

1. General

When pumping power is lost to the primary process pumps supplying the reactor cooling, the flow decays down to an emergency flow provided by low-lift secondary pumps or by a tertiary pumping system. The process flow decays immediately along a curve determined by the pump flywheel inertia and speed and the reactor resistance characteristics. At the same time, this loss of power causes a scram of the vertical safety rods and after an initial delay of one to three seconds, the reactor power level decays more rapidly than the flywheel decay of the reactor flow. It is expected that two to four of the low-lift pumps will continue full level pumping and will take over automatically from the flywheel decay of the high-lift pumps within 40 to 60 seconds. There is a remote chance, however, that this secondary system may fail to respond to the power failure of the high-lift pumps and that the reactor cooling must be provided by the third pumping system.

At this point the flywheel decay flow is tapering off towards zero, while the reactor heat extraction requirements are leveling off to that produced by

* Present expectation is two low lifts plus the standby of a high lift pump. After Design Change #772, it will be four low lifts.

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radioactive decay of the fission products and the graphite sensible heat. Consequently, if this tertiary or last-ditch supply is too late or too little excessive outlet temperatures will be obtained with the possibility of tube vapor lock and fuel burnout resulting. It should be noted, that under these assumed emergency conditions the front riser pressures during last-ditch cooling are very low and cannot be readily increased. Therefore, any flow reduction sufficient to cause tube boiling and vapor lock can result in no flow for perhaps several hours, with the probability of severe fuel damage. Therefore, it is important to have sufficient last-ditch cooling to avoid even temporary boiling.

This third coolant system for the K Reactors consists of an independent diesel-powered pumping station located between the two reactor areas. The pump station is sized to supply either reactor with required emergency coolant flows upon demand but not both reactors simultaneously. The reactor which requires the last-ditch coolant is selected automatically by a hydraulic balance between the two reactors. It is assumed that one reactor will retain its secondary system consisting of two low-lift pumps and the steam turbine pump, which has the capacity of a low lift. The back pressure established at this non-stricken reactor then forces the majority of the diesel pump output to the reactor with no other source of coolant supply. This system is described in detail in HW-74095 VOL3, pages 46-49, (3)

The purpose of the last-ditch coolant system is to provide emergency reactor coolant flow to one stricken K Reactor should, following a local or plant-wide loss of electrical power, the steam-powered secondary coolant system fail to provide adequate reactor cooling. During the power-loss flow transient the reactor coolant flow decreases at a rate determined by the coast-down of the primary coolant pumps until, if the secondary coolant system does not respond, the water pressure in the reactor inlet piping reaches a value equivalent to the pressure output of the accelerated diesel-driven, last-ditch pumps. At this point, the check valve opens in the high pressure crosstie line and last-ditch flow is supplied to the stricken reactor.

2. Measured Reactor Last-Ditch Flow

The last-ditch flow to a K Reactor consists of that portion of the flow from two or three diesel pumps that diverts to the stricken K Reactor rather than flowing to the other K Reactor which has some low-lift pumps in operation. This flow is measured in two manners:

- a. Flow tube meter in the crosstie line to the stricken reactor.
- b. A calibration curve of the bottom of A and B riser pressures (BORP) versus the reactor flow supply through the A and B risers only.

The best available riser calibration curve for the K Reactors is presented in Figures 1 and 2, and is obtained from the experimental data of three recent K production tests, (4,5,6) plus a theoretical riser pressure curve obtained from a system hydraulic study, (7)

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FIGURE 1

11E REACTOR RISE FLOW

LEGEND

		HW-72699 "Reactor Pressure 11E"
		SS Jones
-Temp.-		
17 C	●	HW-74321, PT-II-48
	○*	D. L. Henberg (1-11-66) (4)
5 C	▲	HW-77378, PT-II-57
	△*	L. C. Lesser (2-11-66) (5)
14 C	■	HW-80533, PT-IP-823
	□*	W. H. Radtke (11-01-66) (6)
(* Flow Through A+B Risers only)		

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FIGURE 2
KW REACTOR LOW FLOW

		LEGEND	
—	*	HW-72699 "Reactor Process Flow" SS Jones	
17 C	●	HW-74621, PT-IP-499	
	○*	D. L. Bonheron (5-21-60) (4)	
5 C	▲	HW-77376, PT-IP-573	(5)
	△*	L. C. Lessor (2-13-60)	
14 C	■	HW-80553, PT-IP-625	(6)
	□*	W. H. Padtke (11-01-60)	
(* Flow Through A&B Riser only)			

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The calibration curve for the crosstie orifices is available from the K-Reactor process engineer; who, also, should have the most up-to-date reactor riser calibration curve similar to Figures 1 and 2.

3. Assured Reactor Flow

Because of the extreme importance of adequate last-ditch cooling, as noted earlier, a number of factors are applied to this measured last-ditch flow before it can safely be called assured last-ditch flow.

The general equation to convert from measured to corrected, or assured, reactor flow is:

$$G_A = \frac{F_S F_C F_T}{1 + F_G} G_M, \quad (1)$$

where F_S = the allowance for strainer plugging during an actual emergency,

F_C = the corrosion allowance which depends upon time since line cleaning or actual emergency flow measurement,

F_T = a temperature correction factor to allow for the decreased reactor resistance under the elevated effluent temperatures of emergency conditions, and

$1 + F_G$ = a term to allow for increasing errors in flow measurement at lower and lower flows.

The value and significance of these factors is discussed as follows:

a. Strainer Plugging Factor (F_S)

When detailed last-ditch studies were commenced several years ago, a considerable amount of corrosion products, etc., were observed during crosstie line flushing procedures. There is continued evidence of line corrosion and conservatively a strainer plugging factor of 0.96 is recommended for the possible plugged strainer decrease in diesel pump flow during an emergency event. This represents the coverage of 40 per cent of the strainer area and is considered a conservative but prudent restriction.

There are several important considerations concerning this factor.

1. The severe blocking of only two central zone crossheader screens during an emergency condition could be enough to cause serious fuel burnout and raise serious questions concerning the startup of such a reactor.

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2. There are no strainers in the crosstie line and the emergency flow passes through only the crossheader strainers. Experience has shown that these strainers tend to plug unevenly and therefore individual crossheader flow can be significantly reduced with no apparent change in reactor flow. Therefore, the value of F_S for the total reactor is considered to be 1.0 based upon the requirement that individual crossheader flow reduction must not be excessive.
3. There must be a strainer plugging factor for the individual tubes, however; and since this limit occurs near the top of the reactor where the strainer plugging is the least, a factor of 0.98 is recommended.
4. This 0.98 is an arbitrary value subject to change upon adequate evaluation and notification of the experimental values of strainer plugging obtained during the various reactor tests.

b. Line Corrosion Factor

Considerable crosstie line corrosion was found to have occurred during the system testing upon the completion of Project CGI-844, and the crosstie lines were mechanically cleaned in January, 1963. Their flow was checked at that time and then rechecked in November of 1963. The change in line pressure drop converted to a relative decrease in flow is shown in Figure 3. This was obtained from a report which discusses the significance of tests performed to evaluate the adequacy of the K last-ditch cooling system, wherein the test results were summarized to obtain an average corrosion effect.⁽⁸⁾

The accurate way to include this corrosion and other factors is to calculate the diesel pump supply pressure curve in terms of BORP. Then, the actual reactor flow will be at the intersection of this curve with the reactor demand curve shown in Figures 1 and 2. In view of the low accuracy of the experimental data, however, the considerable additional work involved in this more accurate procedure is not warranted.

c. Temperature Factor (F_T)

Increasing the coolant temperature increases the flow rate. The magnitude of this effect is shown in Figure 4, where the relative flow factor is shown as a function of the inlet coolant temperature. This is based upon five general conclusions.

1. The average coolant temperature in going through the reactor during emergency conditions (90 C bulk outlet) is very close to 55 C.
2. Between test and emergency conditions there would be an increase in reactor flow because of the increase in this average reactor temperatures from 10 to 25 C up to 55 C.

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3. The effect of a coolant temperature change upon the line frictional pressure drop is given by the expression:

$$\Delta P_{f_T} = F_T^* \Delta P_{f_{55}}, \quad (2)$$

where ΔP_{f_T} = the line pressure drop at T °C,

$\Delta P_{f_{55}}$ = the line pressure drop at 55 C, and

F_T^* = the fractional increase in line pressure drop as the coolant temperature is decreased from 55 C.

4. Approximately two-thirds of the flow resistance occurs in the crosstie line with the balance in the reactor. Thus, to determine the effect of reactor emergency temperatures upon the measured cold flow, we can only consider the reactor part of the system. The fractional flow increase for this effect is given as:

$$F_T = \left(\frac{\Delta P_{f_c} + F_T^* \Delta P_{f_R}}{\Delta P_{f_c} + \Delta P_{f_R}} \right)^{.54}, \quad (3)$$

where F_T = fractional increase in reactor flow between test and emergency conditions (i.e., 55 C),

ΔP_{f_c} = crosstie pressure drop - psi, and

ΔP_{f_R} = reactor pressure drop - psi.

5. To compare one measured flow condition with another measured flow at a different temperature, use is made of equation (2) since the whole system is at the given temperature. Thus, to determine the expected measured flow at 25 C from a value measured at 5 C, we use the relationship:

$$G_{25} = \left(\frac{F_{25}^*}{F_5^*} \right)^{.54} G_5 = 1.08 G_5, \quad (4)$$

and values of this exponential pressure drop correction ratio are presented as F_T' in Figure 4. Here F_T' is based upon a 20 C coolant temperature. Applying this to the above problem gives:

$$G_{25} = \frac{F_{25}'}{F_5'} G_5 = 1.08 G_5.$$

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FIGURE 4 - FLOW TEMPERATURE FACTORS

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d. Low Flow Error Factor ($1 + F_G$)

It can be shown that the flow error is related to the various errors in pressure measurement, by the expression

$$\frac{\Delta G}{G} = 0.54 \frac{\Delta P_T + \Delta P_D}{P_T - P_D} ; \quad (5)$$

where $\frac{\Delta G}{G}$ = the fractional error in the reactor flow,

P_T = reactor riser pressure - psig, and

P_D = reactor top of downcomer pressure - psig.

Assuming pressure errors of 1.5 psi riser pressure and 1 psi top of downcomer pressure, this latter made of barometric pressure variations, and actual pressure errors, we obtain for 10,000 gpm flow

$$\frac{\Delta G}{G} = 0.54 \left(\frac{1.5 + 1}{19 - 0} \right) = 0.072. \quad (6)$$

This and corresponding error values are shown as the higher curve in Figure 5.

As noted in a recent report⁽⁸⁾, a survey of the experimental results of measured K-Reactor emergency flow does not indicate high accuracy. Whether this is because of pressure errors, orifice flow meter errors, or pump discharge errors is not known. But the scatter in the experimental results indicates a variety of errors, without any definite trend or evident source, and the higher value of $1 + F_G$ should be used. This should apply until more consistent experimental results warrant an evaluation of lower errors in emergency flow measurement. When this is done, the lower error curve should not be used for test data more than two years old or for flow changes without test data in excess of ten per cent.

With these various factors determined, we can then calculate an assured reactor emergency flow and determine the critical time of emergency flow takeover, as noted in the following section.

4. Critical Flow Time and Heat Extraction Factor

a. Critical Flow and Time

The basic criterion for this condition is the point of maximum bulk outlet temperature during the period of diesel-pump takeover from the flow decay provided by the pump flywheels. Because of the hazard involved,

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this condition is not measured experimentally. However, it can be closely approximated by combining the reactor heat extraction requirements with the experimental flow decay measurements summarized in earlier reports. (4,5) This determination is shown in Figure 6. It occurs just at the point where stable last-ditch flow is achieved.

Because of the reactor shutdown requirements and other difficulties involved in the actual measurement of this critical diesel-pump flow to the reactors, it is often desirable to measure a steady-state diesel-pump flow to one or both reactors. It is even planned to measure changes in diesel-pump flow through a fixed drain line on an intermittent basis. In this case, this flow must be translated to a reactor flow and then to a critical flow time.

Studies are in progress on the comparison of theoretical and actual reactor flywheel flow decay. Until these are completed, an approximate procedure is presented in the specification. This uses the basic flywheel flow decay equation and experimentally determines the lumped parameter α that includes the flywheel inertia, speed, and other flow decay effects.

The basic pump flywheel decay curve can be represented by the simplified expression:

$$f_G = \frac{G_M}{G_{M_0}} = \frac{1}{1 + \alpha \tau} \quad (7)$$

Where G_M = flywheel decay flow seconds after power loss - 1000 gpm,

G_{M_0} = original pumping flow - 1000 gpm,

τ = time after loss of pumps - sec.,

α = over-all flywheel inertia and system constant, and

f_G = fraction of original flow.

This equation was applied to the six-pump flywheel decay, Test A-1 of reference (5), and an average value of .049 was obtained for α .

The typical reactor flow and power-level decay following the loss of pumping power and a reactor scram are depicted in Figure 6. At the same time, the corresponding bulk outlet temperatures are shown for various values of inlet coolant temperature. The significant condition in this figure is that the peak bulk outlet temperatures are obtained just at the point of achieving stable flow, i.e., where the system taking over from the pump flywheel decay has just barely completed the process and stable secondary or last-ditch flow is obtained.

To change from a known to an unknown critical flow-time condition, the following procedure is applied. From the known test conditions, we can determine:

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FIGURE 6

DETERMINATION OF TYPICAL CRITICAL FLOW AND TIME

1. f_{G_o} = the fraction of original full-level flow at the point of stable secondary or last-ditch flow during the pump flywheel decay,
2. τ_{c_o} = the time that this stable flow is just achieved - seconds, and
3. τ_o = the theoretical time that this flow would be achieved from the relationship:

$$\tau_o = \frac{1 - f_{G_o}}{\alpha f_{G_o}} = \frac{1 - f_{G_o}}{(.049) f_{G_o}} \quad *$$
(8)

This implies, and correctly so, that a deviation exists between the theory and experiment. This is small and using the theory to change the experimental results on a ratio basis should provide a fairly accurate new value of critical flow time. Thus, for the case of a new emergency flow, we first determine**:

f_{G_x} = the ratio of the stable emergency flow to the full level flow.

Then we calculate the theoretical critical flow time according to the equation.

$$\tau_x = \frac{1 - f_{G_x}}{(.049) f_{G_x}} \quad ,$$
(8a)

and finally we determine an empirical value of the expected critical flow time for the new conditions according to the ratio expression

$$\tau_{c_x} = \tau_{c_o} \left(\frac{\tau_x}{\tau_o} \right)$$
(9)

Where τ_{c_x} = the actual critical flow time for the condition of the assured stable flow,

τ_x = the theoretical critical flow time, and

subscripts o and x refer to the test flow decay time and the unknown flow decay time, respectively.

* Where α has been determined from one decay test, reference (5) Test A-1, and is subject to change upon future testing.

** A simple approach to this problem was developed by W. H. Radtke, namely that a linear plot of f_G vs. f_r is obtained. This bypasses the determination of the critical time and should apply for all cases where significant changes have not been made in reactor flow or primary pumping characteristics. The linear curve is presented in Figure 7.

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FIGURE 7

HEAT EXTRACTION FACTOR FOR VARIOUS
LAST-DITCH CRITICAL FLOWS

Heat Extraction Factor

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b. Heat Extraction Requirements after Emergency Scram (f_T)

The amount of coolant required is a direct function of the amount of heat to be extracted and this in turn depends upon the temperature rise of the coolant in passing through the reactor. In a recent test, the effect of raising the coolant outlet temperature was investigated. On the basis of this test, a modified heat output curve was obtained.⁽⁹⁾ For convenience, this is presented in Figure 8 as it applies to the K Reactors.

5. Maximum Reactor Power Level for Adequate Last-Ditch Cooling

a. Outlet Temperature Limit

1) Reactor Emergency Bulk Temperature Limit

During emergency flow, bulk boiling may occur in the reactor effluent system at the top of the downcomer at temperatures as low as 98 C. At these low emergency flows, there is very little pressure driving force and bulk boiling with the accompanying pressurization could readily lead to fuel burnout and serious reactor damage. Therefore, the maximum allowable bulk outlet temperature at 20 or more seconds following a reactor scram is established as 90 C. This is presented in Technical Bases D-1.⁽¹⁰⁾

2) Reactor Emergency Tube Outlet Temperature Limit

Laboratory studies of individual tubes at the low pressures associated with emergency flow conditions have indicated that fuel burnout can occur at shutdown power level conditions and that cooling by boiling cannot be depended upon to ensure adequate fuel cooling.^(11,12) Therefore, it is important that tube outlet temperatures be limited to values below the saturation point. Conservatively, a limit has been established of 5 C below the corresponding rear crossheader saturation temperature. This saturation temperature, and thus the tube outlet temperature, increases as we go down the reactor, by as much as 10 C. Thus, from these considerations, it is best to peak the reactor power level at the bottom rather than the top of the unit.

The tube outlet emergency temperature limit, T_L , (equal to crossheader saturation temperature minus 5 C) depends upon the crossheader pressure. This is determined as the sum of the static and friction pressures from the top of the downcomer down to the first headers with maximum powered tubes. It is assumed that the top of the downcomer is at atmospheric pressure. This value of 5° allows for a cumulative error of four feet in the location of the highest temperature crossheader, one foot in the level of water in the downcomer approach section, 1 psi error in the frictional pressure drop in this section of the effluent system, and 0.4 psi depression of the top of downcomer pressure below 14.7 psia. This latter effect being a combination of small suction pressure effects created by the downcomer system, and barometric changes. Based upon these assumptions, the limiting tube outlet temperatures at 20 seconds or more after a reactor loss of pumping power are shown in Figure 9.

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Fraction of Original
Heat Extraction, f_T

FIGURE 8
KE, KW REACTORS RELATIVE HEAT EXTRACTION
REQUIREMENTS FOLLOWING A REACTOR SCRAM

(Reference 9)

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FIGURE 9

K REACTORS EMERGENCY FLOW
TUBE OUTLET TEMPERATURE LIMIT

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b. Power Level Adequacy

1) Bulk Temperature Limit

The basic equation for the determination of the maximum reactor power level, for which adequate last-ditch cooling is assured, is given as:

$$P = \frac{0.264}{f_{\tau}} G_A (90 - T_{i_c}); \quad (10)$$

where P = the maximum power level for cooling adequacy based upon the bulk outlet conditions - Mw,

G_A = assured diesel-pump flow to the stricken K Reactor - 1000 gpm,

T_{i_c} = the crosstie line temperature - °C,* and

f_{τ} = the heat extraction decay factor from Figure 7 or 8 at the time of diesel-pump critical flow.

2) Tube Outlet Temperature Limit

As noted earlier, we have both a bulk and a tube temperature limit. The latter is given by the relationship:

$$P_t = \frac{0.264}{f_{\tau}} F_S G_A (T_L - T_{i_c}) F_{\epsilon}; \quad (11)$$

where P_t = the maximum power level for cooling adequacy based upon tube outlet temperature limits (i.e., crossheader saturation temperature) - Mw,

F_S = crossheader strainer plugging factor assumed = .98,

G_A = the assured diesel-pump critical flow in 1000 gpm,

T_L = the tube outlet temperature limit - °C (Values of this limit for the K Reactors are presented in Figure 9.),

T_{i_c} = the crosstie coolant temperature - °C,

F_{ϵ} = the reactor flattening efficiency = .84 as of January, 1964, and

f_{τ} = the heat extraction decay factor from Figure 7 or 8 at the time of critical diesel-pump takeover.

* This is now arbitrarily assumed to be 5 C above process water inlet temperature.

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With the power level determined for both bulk and the tube temperature limits, the lower of the two levels should be used. In general above 5-10 C, the bulk temperature limit is always controlling and often it is controlling over the whole range of coolant inlet temperatures.

6. Required Last-Ditch Cooling for a Desired Reactor Power Level

In general, the foregoing process is used in reverse. There are however several important points to be noted. These will be discussed, briefly summarizing the actual process of the determination of flow requirements.

a. Required Assured Last-Ditch Flow

This is readily obtained by transposing equations (1), (8), and (9).

Limiting Bulk Outlet Temperature

$$G_{AB} = \frac{f \tau P_o}{0.264 (90 - T_{ic})} , \quad (10a)$$

$$G_{MB} = \frac{(1 + F_G) G_{AB}}{F_T} ; \quad (1a)$$

where G_{MB} is the value of diesel-pump flow (to the stricken reactor) that should be measured to assure adequate bulk cooling at the desired reactor power level.

Limiting Tube Outlet Temperature*

$$G_{At} = \frac{f \tau P_o}{0.264 (T_L - T_{ic}) F_e F_S} , \quad (11a)$$

$$G_{Mt} = \frac{(1 + F_G) G_{At}}{F_T} ; \quad (1b)$$

where G_{Mt} is the value of diesel-pump flow to the stricken reactor that should be measured to assure adequate process tube cooling at the desired reactor power level. The important process here is to use both equations, i.e., 10a and 1a or 11a and 1b in determining the flow.

7. Corrosion and Pumping Effects

The current crosstie pipe line corrosion conditions are shown in Figure 3 and explained in detail in reference (8). This is a situation that will require

* The strainer plugging factor F_S is arbitrarily placed in equation (11a) rather than (1) to yield a constant ratio of assured to measured flow for both bulk and tube considerations.

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close following and is subject to considerable change as steps are taken to reduce or eliminate the problem. It will be important to update the correction factors presented in Figure 3 as changes are made and additional test results obtained.

a. Effect of Pumping Conditions at Other Reactor

The emergency flow diesel pumps provide flow and pressure to the junction of the diesel pump discharge line with the high pressure crosstie. From here the flow is distributed to the KE and KW Reactors according to their riser back-pressure and their crosstie hydraulic resistance. The greater the secondary system flow to the unstricken reactor, the greater the diversion of emergency flow to the stricken unit. In the use of this analysis, however, it is only important to know the stricken reactor's critical flow. The same method for the determination of last-ditch cooling adequacy should be used irrespective of the number of pumps in operation at the aiding reactor. This number will, of course, influence the measured flow value (G_M) to be used for the determination of last-ditch cooling adequacy.

This completes the outline of a method for determining the cooling adequacy of the K-Reactor last-ditch cooling system. The method can be used with minor modifications for determining the cooling adequacy of the secondary cooling system. In general, however, the capacity of this system is so much in excess of its requirements that the various flow correction factors can be omitted.

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REFERENCES

- (1) Trumble, R. E. Performance Criteria for the Hanford Production Reactors, HW-66929. September 1, 1960.
- (2) Van Wormer, F. W. Performance Criteria for Irradiation Processing Department Production Reactors, HW-79695. November 20, 1963.
- (3) Staff of Irradiation Processing Department. Hazards Summary Report - Volume 3 - Description of the 100-KE and the 100-KW Production Reactor Plants, HW-74095 VOL 3 April 22, 1963. pp. 46-49. (SECRET).
- (4) Renberger, D. L. Numerical Results of PT-IP-499-AM, Test of the 182-K Diesel Pumps at KE and KW, HW-74621. August 21, 1962. (SECRET).
- (5) Lessor, L. C. Pressure and Flow Data PT-IP-573-I, 100-K Flow Tests, HW-77378. April 24, 1963. (SECRET).
- (6) Radtke, W. H. Results of Production Test IP-628-I, Crosstie Flow Test and Flushing Calibrations, HW-80533. January 23, 1964. (SECRET).
- (7) Jones, S. S., J. E. Morrissey, and J. L. Manuel. Reactor Process Flow Technical Data, HW-72699. February 15, 1962.
- (8) Jones, S. S. Technical Note on the Results of Production Test IP-628-AI, Crosstie Flow Test and Flushing Calibrations, HW-81027. February 27, 1964.
- (9) Jones, S. S. and J. L. Manuel. Numerical Results of PT-IP-338-A SUPP A, DR-Reactor Heat Decay Test at High Outlet Water Temperatures, HW-75727. December 3, 1962. (SECRET).
- (10) Jones, S. S. Technical Bases for Standards, HW-55500 RD. January, 1964.
- (11) Waters, E. D. and D. E. Fitzsimmons. Pressure Required to Overcome Tube Burnout During Transient Conditions, HW-66274. August 5, 1960. (SECRET).
- (12) Carlson, P. A. and S. S. Jones. Potential for Process Tube Burnout During Transient Conditions, HW-61760. April 22, 1960. (SECRET).

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