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TECHNICAL NOTE ON RESULTS OF PRODUCTION TEST IP-628-AI; CROSSTIE FLOW TEST AND FLUSHING CALIBRATION, AND THE EARLIER PRODUCTION TESTS IP-573-AI and IP-499-AI

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TECHNICAL NOTE ON

RESULTS OF PRODUCTION TEST IP-628-AI:

CROSSTIE FLOW TEST AND FLUSHING CALIBRATION

AND THE EARLIER PRODUCTION TESTS

IP-573-AI AND IP-499-AI

by

Stephen S. Jones

June 15, 1964

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TECHNICAL NOTE ON RESULTS OF PRODUCTION TEST IP-628-AI:

CROSSTIE FLOW TEST AND FLUSHING CALIBRATION

AND THE EARLIER PRODUCTION TESTS IP-573-AI AND IP-499-AI

INTRODUCTION

This report reviews the numerical results presented in the last three production tests that have been performed on the K-Reactor emergency cooling. Their results are combined to obtain flow error factors, compared to obtain line corrosion effects, and analyzed to determine the degree of last-ditch cooling adequacy. In addition to the actual test results this report also considers the effects of increased diesel pump speed and the reliability and cooling adequacy improvements expected upon the completion of Design Change No. 772. Furthermore, the Reactor Engineering responsibility, noted in the three production tests, of interpreting the degree of cooling adequacy is provided herein.

The importance of this work is evidenced by the \$2,100,000 recently spent under Project CGI-844 to change this last-ditch cooling system from a very inadequate and unreliable system to a system with adequate and reliable last-ditch cooling. The results obtained showed borderline adequacy and this problem will remain important under a reasonable degree of excess adequacy (~10%) is attained.

The numerical results presented in the production test reports were used to obtain the basic last-ditch system conditions of:

Diesel pump discharge pressures,

Pump manifold discharge pressures,

Crosstie line corrosion effects on emergency crosstie flow,

and other significant factors.

These results are applied to the development of a program of relatively minor expense that should achieve the desired cooling adequacy from this last-ditch system.

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SUMMARY

A series of three experimental tests was conducted on the operation and adequacy of the K-Reactors' third, or last-ditch, cooling system. The first test showed considerable line corrosion, the second test was performed directly after line cleaning, and the third test showed a significant amount of additional line corrosion after only nine months of service. The present cooling adequacy of this last-ditch system at the KE and KW reactors is summarized in Figures 1 and 2. These show the power levels for which we have adequate last-ditch cooling as a function of the crosstie coolant temperature. These figures include the effects of increasing the number of pumps that remain in operation at the other K-Reactor, and various other operating or emergency conditions. These curves are for diesel pump speeds of 750 rpm which are planned for June of this year. In these figures the crosstie temperature is assumed to be a conservative 5°C above the process water inlet temperature.

DISCUSSION

1. General

Two important K-Reactor projects were completed in 1962. The process pumps and motors were modified to yield a reactor flow increase of about 15 per cent,⁽¹⁾ and the reactor last-ditch cooling system was modified by the installation of a new diesel-powered pumping station and other improvements.⁽²⁾

After completing these projects a series of tests was conducted in July, 1962, to ascertain the adequacy of this modified last-ditch cooling system. The test results and line inspection made at that time showed inadequate last-ditch system capacity and appreciable line corrosion.⁽³⁾ The crosstie lines were cleaned early in 1963 and two additional system tests have been conducted, one immediately after line cleaning and one nine months later.^(4, 5) In order to obtain the best interpretation of the last test, performed to indicate our current last-ditch cooling adequacy, the results of all three tests are analyzed.

The basis for this analysis is presented in a technical note⁽⁶⁾ which describes the recommended procedure for determining:

- a. the maximum adequate operating power level for a given measured emergency coolant flow and temperature, and
- b. the required coolant flow and temperature to achieve adequate emergency cooling from a desired operating power level.

In this report we are concerned with method a, the determination of the reactor power level for which we have adequate last-ditch cooling. The various steps involved in this determination are noted in the following sections.

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FIGURE 1
KE-REACTOR POWER LEVELS FOR ADEQUATE LAST-DITCH COOLING
(2 Diesel Pumps, 750 rpm)

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FIGURE 2
KW-REACTOR POWER LEVELS FOR ADEQUATE LAST-DITCH
COOLING (2 Diesel Pumps, 750 rpm)

2. Assured Emergency Flow

Because of the essential nature of reactor cooling, a number of factors are applied to the measured reactor flow to obtain an assured flow. This is a conservative value because each factor is considered independently and taken in a conservative manner. However, the statistical approach to this value is not justified by the very high element of risk involved.

The equation presented in Reference 6 to convert from measured to assured flow is:

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

where

G_A = Assured emergency flow to a stricken K-Reactor, 1000 gpm

F_C = Crosstie line corrosion factor, i.e., fraction of clean line flow

F_T = Flow temperature factor fraction flow increase between test and emergency conditions

F_G = Low flow error factor fraction of flow that may be in error

G_M = Measured stable emergency flow, 1000 gpm

These various factors are described in Reference 6 and recommended values are presented. At the same time the values of F_C and $1 + F_G$ in this reference were influenced by the experimental results summarized in this report. Thus, there is a mutual dependence between this report and the analysis presented in Reference 6. This is made apparent in the following sections.

a. Crosstie Line Corrosion

In this system the corrosion effect of concern is the decrease in flow that it produces. This effect is determined by measuring the pressure drop in the crosstie lines for various flows and at various times after line cleaning.

The details of this determination are presented in Appendices B and C of this report. Appendix B presents the determination of the diesel pump discharge pressures and the pressures at the junction of the diesel pump discharge manifold with the crosstie lines to each reactor. Appendix C presents the experimental corrosion effects on the crosstie line pressure drop. From this analysis a corrosion flow factor F_C was determined. The change in this

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factor with corrosion time, based upon the experimental data, is summarized in Table A-II and shown graphically in Figure 3.

b. Low Flow Accuracy

The flow to the so-called "stricker" reactor, i.e., the reactor receiving only diesel pump flow, was measured in two manners:

- 1) Careful determination of the bottom of front, A and B riser pressure (BORP), and the conversion of this pressure to flow by means of the combined theoretical and experimental curves presented in Figures A-1 and A-2.
- 2) Actual measurement of the pressure drop across the crosstie flow-tube and conversion to flow by means of the theoretical flow-tube calibration curve.

In comparing the flows obtained by these two methods there was an inconsistent scatter of the flow values that could be interpreted only as an indication of significant flow measurement error. These values are presented in Table I and lead to the recommendation that the higher of the two $1 + F_G$ error curves in Figure 5 of Reference 6 be used until more accurate low flow measurements can be confirmed. Thus, using the values for the F_T , F_C , and $(1 + F_G)$ factors presented in Reference 6 we can calculate the assured reactor flow for any given measured last-ditch flow. Furthermore, using the F_T relationship shown in Reference 6 and a value of the corrosion factor F_C , from Figure 3, we can determine the assured reactor flow for any crosstie water temperature and time after line cleaning. This has been done for the K Reactors for February and July of 1964, for different amounts of last-ditch and secondary-system pumping. The results are summarized in Table II.

3. Critical Last-ditch Flow and Heat Extraction Requirements

The reactor process pumping power is normally provided by the Bonneville Power Authority (BPA) from a network of 230 kv transmission lines. Upon the inadvertent loss of this power the reactor flow decays along a curve determined by the pump flywheel speed and inertia and upon the flow characteristics of the system. The flow would normally level off at that provided by two or three low-lift pumps operated by steam-generated electric power. In the very unlikely event of the failure of this secondary system, the flywheel decay flow would level off at that provided by the emergency diesel pumps. These start up within four seconds following the loss of BPA power and provide emergency flow to the crosstie lines leading to each K Reactor. A hydraulic balance is established at this junction of the diesel discharge manifold with the crosstie lines. The sum of the reactor A and B riser pressures plus the frictional flow pressure drop from the risers to this junction must be the same for each

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TABLE I

EXPERIMENTAL LOW FLOW DISCREPANCIES

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Test	Reactor	Riser Pressure (BORP) psig.	Crosstie Flow		% Flow Discrepancy**
			From Riser Pressure*	From Flow- Tube	
			1000 gpm		
PT-IP-499	KW				
5.4.5-c		40.1	27.5	27.6	.4
d		41.0	28.4	28.2	.7
e		43.0	30.8	30.0	2.6
f		45.3	33.3	32.0	4.0
f-1		42.0	29.4	29.0	1.4
g		39.0	26.3	26.5	.8
h		45.4	33.2	29.9	11.0
i		41.4	29.0	28.7	1.0
j-1		39.8	25.7	28.1	8.5
k		35.3	21.7	22.2	2.3
l		38.2	25.3	25.5	.8
n		44.8	32.5	31.3	3.8
p		34.8	20.8	22.0	5.5
q		42.8	30.7	29.9	2.7
5.4.6-c	KE	38.7	25.7	26.7	3.7
g		37.4	24.3	25.4	4.3
h		41.4	29.2	28.9	1.0
i		40.8	28.2	27.6	2.2
h-1		42.3	30.4	29.4	3.4
l	KW	37.1	24.0	23.8	0.8
5.4.7-f		40.9	28.6	27.4	4.4
5.4.7.1-f		41.2	28.7	28.1	2.1
5.4.8-f		KE	40.2	27.6	26.7
5.4.8.1-f	40.8		28.5	26.5	7.5
PT-IP-573	KE				
B- 1		43.2	30.5	30.9	0.6
2	KW	44.3	32.0	31.9	0.3
3		42.2	32.3	33.0	2.1
3'		43.5	34.0	32.2	5.6
3''		41.2	31.3	32.5	2.5
6		44.4	32.3	31.2	3.6
7		41.1	28.3	29.8	5.0
8		47.9	35.8	33.8	5.9
11		51.3	39.5	35.9	10.0
14		46.5	34.2	31.6	8.3
14'		46.6	34.3	31.5	8.8
PT-IP-628	KE				
A- 1		40.5	28.0	28.9	5.0
2	KW	38.7	25.8	27.1	4.8
3		41.5	29.0	29.8	2.7
3'		43.7	31.4	29.9	5.0
4		44.8	32.5	30.3	6.8

*From Figures 1 and 2 of Reference 6

**ΔFlow ÷ Flow-tube flow

TABLE II

ASSURED REACTOR EMERGENCY FLOW

TEST		AIDING REACTOR		DIESEL PUMPS *			STRICKEN REACTOR									
PT No.	Test No.	Unit	No. of L. L. Pumps	No. Oper.	Total Flow 1000 gpm	Fract. of Flow to Str. Reactor F_D	Unit	Cross-tie Temp. T_{iC}	Meas. Emerg. Flow 1000 gpm G_M	Corrosion		Temp. Cold F_T'/F_T' (Test)	Fact. Emerg. F_T	Low Flow Error Fact. $1 + F_G$	Assur. Emerg. Flow 1000 g	
										Years Since Clean. X	Corr. Fact. F_C				G_A	Avg. Simil. Test
28	A- 2	KW	2	2	39.7	.68	KE	14	27.1	.7	1.0	1.0	1.04	1.06	26.6	27.3 27.5
								5		1.0	.98	.96	1.06		25.5	
								20		1.5	.95	1.02	1.03		25.5	
	A- 1		3		37.5	.77		14	28.9	.7	1.0	1.0	1.04		28.4	
								5		1.0	.98	.96	1.06		27.2	
								20		1.5	.95	1.02	1.03		27.2	
73	B- 1		3		40.2	.77		5	30.9	0	1.0	1.0	1.06	1.05	31.2	
								5		1.0	.88	1.0	1.06		27.4	
								20		1.5	.86	1.07	1.03		27.8	
	B- 2		4		39.3	.81		5	31.9	0	1.0	1.0	1.06		32.2	
								5		1.0	.88	1.0	1.06		28.4	
								20		1.5	.86	1.07	1.03		28.6	
573	B- 7	KE	2	2	42.8	.70	KW	5	29.8	0	1.0	1.0	1.06	1.06	29.8	28.8 29.2
								5		1.0	.93	1.0	1.06		27.7	
								20		1.5	.91	1.07	1.03		28.1	
	B-14		3		40.5	.78		5	31.6	0	1.0	1.0	1.06		31.6	
								5		1.0	.93	1.0	1.06		29.4	
								20		1.5	.91	1.07	1.03		29.8	
628	A- 3		3		37.8	.79		14	29.8	.7	1.0	1.0	1.04		29.2	29.6 29.7
								5		1.0	.99	.96	1.06		28.3	
								20		1.5	.97	1.02	1.03		28.6	
	A- 4		4		37.0	.82		14	30.3	.7	1.0	1.0	1.04		29.8	
								5		1.0	.99	.96	1.06		28.8	
								20		1.5	.97	1.02	1.03		29.1	
573	B- 3'		4		39.4	.82		5	32.3	0	1.0	1.0	1.06		32.5	
								5		1.0	.93	1.0	1.06		30.3	
								20		1.5	.91	1.07	1.03		30.6	
	B- 8		2	3	54.8	.62		5	33.8	0	1.0	1.0	1.06	1.05	34.1	
								5		1.0	.93	1.0	1.06		31.8	
								20		1.5	.91	1.07	1.03		32.2	

*All test results were at 720 rpm diesel pump motor speed.

reactor. Therefore, the diesel pump flow split is determined as that necessary to yield equal junction pressures; and the more riser pressure that is provided at the aiding reactor* by its secondary pumps, the less diesel pump flow will be diverted to it. Because of the limited capacity of these diesel pumps, it is necessary to minimize this flow diversion to the aiding, or unstricken, reactor.

The critical flow is the stable emergency flow provided by the diesel pumps to the stricken reactor, and the critical time is the earliest time that this stable flow is achieved. At this time, the critical coolant flow must be able to handle the reactor heat extraction requirements. Values of this heat extraction factor are presented in Reference 6, Figure 7, as a function of the critical flow factor F_G , defined as the stable emergency flow divided by the full level operational flow.

Because of the conservative manner in which most other factors in this problem have been handled, a nonconservative but more accurate condition is considered with respect to this critical emergency flow. In practice the assured emergency flow is 3 to 12 per cent less than the measured value. This means that during the pump flywheel decay this assured flow would be reached some time later than the measured value, with a corresponding decrease in the heat extraction factor. Advantage of this is taken in our calculations and the heat extraction f_T is determined from the emergency flow factor F_G that is the assured flow over the original full level flow. Values of f_T versus F_G are presented in Figure 7 of Reference 6.

4. Last-ditch Cooling Adequacy

With the assured reactor critical flow and the heat extraction factor, as determined in the foregoing sections, the maximum power level for adequate emergency cooling is given by the relationship

$$P_B = \frac{0.264 G_A(90 - Ti_c)}{f_T} \quad (2)$$

where

P_B = Maximum power level for adequate last-ditch cooling - Mw,
i.e., bulk outlet temperature $\leq 90^\circ\text{C}$ **

G_A = Assured last-ditch flow, 1000 gpm

Ti_c = Crosstie coolant temperature - $^\circ\text{C}$

f_T = Heat extraction factor at critical takeover time

*Throughout this report the term "aiding reactor" refers to the K-Reactor which still has secondary pumping after a complete BPA power failure.

**There is also a tube outlet temperature limit equal to crossheader saturation temperatures minus 5°C . For normal reactor configuration with a flattening efficiency of greater than 0.82 the bulk temperature is more limiting.

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The generalized application of this equation is presented in Figure 4a. This shows the assured flow requirements for various reactor power levels at varying crosstie coolant temperatures. Superimposed upon this figure is a curve of the 95° bulk outlet power level at 205,000 gpm flow and various process water inlet temperatures. Figure 4b is a supplemental chart showing the effect upon the assured coolant flow requirements of:

- a. changing the reactor total flow while maintaining a 95° bulk operating temperature, and
- b. elevation of the crosstie temperature above that of the process water inlet.

This figure is based upon assured reactor flow and therefore does not include the corrosion, temperature, and error factors that relate to the translation of measured to assured flow. It does, however, include the effect of flow changes upon the heat extraction factor, f_t . The use of this figure is best explained by illustrative examples:

Example 1: Determine the assured flow required for an operating power level of 4300 Mw and flow of 210,000 gpm. The crosstie temperature is 3° above the process inlet of 17°C.

From Figure 4a

$$G_A \text{ at } 4300 \text{ Mw and } 17^\circ = 25,100 \text{ gpm}$$

From Figure 4b

$f_Q = 1$ (Bulk temperature limit not specified and for a constant power level the assured flow requirement is essentially constant over a range of total reactor flows from 200,000 to 210,000 gpm.)

$f_t = 1.041$ by interpolation for 17° process water temperature.

Answer

$$G_A' = f_t G_A = 26,200 \text{ gpm}$$

Alternate Method

Figure 4a can be used directly:

$$\text{at } 4300 \text{ Mw and } 20^\circ \text{ temperature, } G_A = 26,400 \text{ gpm}$$

Example 2: Determine the adequate cooling power level, or the per cent excess flow if the power level exceeds the bulk temperature limit, when the assured flow equals 27,500 gpm and the crosstie temperature is 5°C

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above an inlet temperature of 20°C. The normal reactor flow is 202,000 gpm.

From Figure 4a

At 95° bulk, 202,000 gpm and 20°C inlet temperature, $P_B = 4000$ Mw.

At 4000 Mw and 20°C, $G_A = 24,300$ gpm.

From Figure 4b

At a crosstie temperature, 5° above process, and a process temperature of 20° $f_T = 1.076$.

Therefore,

$$G_A' = f_T G_A = 26,400 \text{ gpm required flow.}$$

Answer

Comparing the assured flow of 27,500 with the required value of 26,400 indicates an excess assured flow of 4 per cent.

Figure 4a deals with assured flow only. The important flow value from an operations viewpoint is the flow that must be measured to obtain this assured flow. The conversion from measured to assured flow was presented in section 2. The experimentally-measured flows, corrected for increased line corrosion effects, were used in the determination of reactor power levels for adequate last-ditch cooling. This is given in Table III, and these power levels are shown graphically as a function of crosstie temperatures in Figures 1 and 2.

Based upon Table III and Figures 1 and 2, several conclusions may be drawn concerning the cooling adequacy of the K last-ditch system. These are:

- a. At 750 rpm diesel pump speed with 2 diesel pumps and 2 low-lift pumps at the aiding reactor
 - 1) KE will be inadequate at process water temperatures above 12°C, and
 - 2) KW will be adequate by 5 per cent all year, but will probably become inadequate in 1965.
- b. With 2 diesel pumps and 3 low-lift pumps at the aiding reactor
 - 1) KE will be adequate by 3 per cent all year, but will be inadequate next spring, and
 - 2) KW will be adequate by 8 per cent and may be adequate all next year.

After the completion of Design Change #772, we would expect a minimum last-ditch pumping of 2 diesel pumps and 4 low-lift pumps at the aiding reactor. This should yield an additional 2 per cent flow to the stricken reactor. Experimental confirmation of this condition will be made.

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T A B L E III

LAST-DITCH COOLING POWER LEVEL ADEQUACY

Pumping Conditions		Stricken Reactor Conditions											
		No. Pumps at Aid. Reactor	No. Diesel Pumps	Unit	Flow			Heat Extr. Fact. f_r	Coolant Temp.		Power Level Adequacy		
					Oper. 1000 gpm	Emerg.	f_G		Proc. T_i	Crosstie T_i °C	Cooling Adeq. at Pump Speed		Bulk Limit
											G _D	G _A	
Feb Aug (1964)	2	2	KE	206	25.5	.124	.112	5	10	4810	4950	4900	
					25.5	.124	.112	20	25	3910	4020	4100	
Feb Aug	3	2			27.3	.133	.114	5	10	5050	5200	4900	
					27.5	.136	.114	20	25	4140	4250	4100	
Feb Aug	4	2			28.4	.138	.115	5	10	5200	5350	4900	
					28.6	.139	.115	20	25	4260	4380	4100	
Feb-63* Feb-64 Aug	2	2	KW	209	29.8	.143	.116	5	10	5420	5570	4950	
					27.7	.133	.114	5	10	5120	5270	4950	
					28.1	.134	.114	20	25	4220	4350	4150	
Feb Aug	3	2			28.8	.138	.115	5	10	5280	5430	4950	
					29.2	.140	.115	20	25	4350	4470	4150	
Feb Aug	4	2			29.6	.142	.115	5	10	5410	5570	4950	
					29.8	.143	.116	20	25	4400	4530	4150	
Feb Aug	2	3			31.8	.152	.117	5	10	5730	5900	4950	
					32.3	.155	.118	20	25	4680	4820	4150	

*Cleaned Lines

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- c. Continued line corrosion with a continued decrease in last-ditch system flow is to be expected until this important problem is resolved by line cleaning and coating or temporarily relieved by line cleaning only.

From Table III and the Figures 1 and 2, it is clear that the number of low-lift pumps in operation at the aiding reactor during this emergency event has a considerable bearing on the cooling adequacy of the diesel system. This determination of the number of pumps to consider in operation at the aiding reactor depends upon a number of factors which are not easy to define. The various aspects of this problem are discussed in section 5.

5. Dependence of Cooling Adequacy upon Pumping Conditions

As noted earlier, the emergency diesel pumps tend to supply flow to both reactors. Our problem is to establish the credible degree of pumping at the aiding reactor so that we can, in turn, decide upon the emergency flow to be expected at the stricken reactor. An important aspect of this problem is the requirement of our coolant criteria that "the failure of a single component of our cooling system shall not render that system inadequate".(7, 8) The coolant systems under consideration are defined as the following:

System 1: Secondary pumping system at a K Reactor consisting of 2 low-lift pumps and one steam pump (equivalent to 3 low-lift pumps).

System 2: Similar system at the other K Reactor.

System 3: The three emergency diesel pumps situated of the crosstie lines that provide flow to each reactor.

System 4: The last-ditch system consisting of the secondary low-lift pumps that remain in operation at the unstricken reactor plus the diesel pumps that can be relied upon to assume full pumping pressure following a BPA power outage.

The sequence of events that occurs in these systems followings a BPA failure is summarized in Table IV. The complete loss of primary and secondary power is assumed for one reactor with no additional component failures. Conditions are presented for both present operation and for operation after the completion of Design Change #772.(9)

It is clear from Figures 1 and 2 that when the last-ditch system functions properly, i.e., no component failures, we have no problem of inadequacy. In fact, we have at least 7 per cent excess capacity. As previously noted, however, we prudently require that our system be adequate even after a component failure. In Table V, the effect is presented of various system component failures upon the pumping conditions at the aiding reactor.

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TABLE IV

EVENTS FOLLOWING THE LOSS OF BPA POWER

Sequence of Events After Power Failure	Present Conditions	After Completion of Design Change #772
1. Loss of BPA power.	<p>a. High-lift pumps decay along flywheel decay curves.</p> <p>b. #1 and #6 low-lift pumps lose BPA power.</p> <p>c. #2 and #5 low-lift pumps automatically separate from their busses and stop pumping.</p> <p>d. The low-lift steam pump automatically accelerates from rest to full head within 45 seconds.</p> <p>e. The three 182-Bldg. diesel pumps accelerate to full head within 5 seconds.</p>	<p>b. #1 and #6 low-lift pumps continue under emergency power.</p> <p>d. The steam pump does not start, but #3 and #4 low-lift pumps continue in operation.</p>
2. Stable secondary pumping flow is achieved at the aiding K-Reactor within 80 seconds.	From: 2 low-lift pumps and 1 steam pump at aiding reactor.	From: 4 low-lift pumps at aiding reactor.
3. Stable last-ditch pumping flow achieved at stricken reactor.	Stable flow from three diesel pumps within 105 seconds.	Stable flow from three diesel pumps within 100 seconds.

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TABLE VTHE EFFECT OF SECONDARY SYSTEM COMPONENT
FAILURES UPON BACKUP ADEQUACY

Failure Event	No. of Remaining Aiding Reactor Pumps		Remarks
	Present Oper.	After D.C. #772	
1. Failure of a secondary pump (i.e., steam or low-lift pump)	2	3	<p>a. Under present operation the steam pump must accelerate to give 3 low-lift pumping flow. This pump could fail to start or could have been inoperative for perhaps 20 days, therefore, its failure cannot be considered a simultaneous unrelated event with respect to the BPA loss.</p> <p>b. After D.C. #772, a low-lift pump that was operating must drop off the line. This appears to be a simultaneous unrelated event and can be incredible.</p>
2. Failure of an electrical bus.	2	2 or 3	<p>a. Under present operation one low-lift pump would drop off with the isolated bus, the steam pump would accelerate and there would be the equivalent of 2 low-lift pumps.</p> <p>b. After D.C. #772, 2 low-lift pumps would drop off with the failed bus; the steam pump would accelerate to give 3 pumps <u>if</u> it were connected to the boiler providing power to the failed bus.</p>
3. Failure of an emergency turbine or generator.	2	2 or 3	The same as items 2a and 2b above.
4. Failure of an emergency boiler.	1 or 2	2	<p>a. Under present operation one low-lift pump would drop off with the failed boiler and the steam pump would automatically accelerate <u>provided</u> it was connected to the unfailed boiler. It would be more likely to have the boiler fail that is supporting the steam pump.</p> <p>b. After D.C. #772, 2 low-lift pumps would be in operation supplied power by the unfailed boiler.</p>

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In addition to the possibility for single component failure in the secondary system of the aiding reactor, we also have the possibility that one of the three diesel pumps will fail to start. The loss of a diesel pump has a greater effect upon adequacy than any other single component of this overall last-ditch system (i.e., system #4). The marked difference in cooling adequacy between two-pump and three-pump diesel operation is shown in Figure 2. The basis for assuming that our "worst credible" incident involves the loss of only one component in our combined diesel pump station and the secondary pumping system at the aiding reactor is presented in a recent document.⁽¹⁰⁾ It is essential to realize that this conclusion requires very high maintenance of the K-Reactors' secondary systems. For, although all of one secondary system may be lost, we must assume the failure of no components in the secondary system that continues in operation.

An important presentation in Table IV is the difference in the degree of back-up pumping between present conditions and post Design Change #772. This yields a marked increase in reliability and an increase in the aiding reactors pumping from three to four pumps.

6. Improving the Last-ditch System's Adequacy

It would be very desirable from both safety and economic considerations to have at least 10 per cent excess capacity in this last-ditch system. This would permit us to reduce our activities in this field to that of routine annual monitoring without the continuous problem that it has been for a number of years. It is believed that such a margin is available through numerous changes, each small in itself but adding to an impressive total of nearly 12 per cent. These various changes are noted below:

a. Crosstie Line Cleaning

The benefits to be expected from line cleaning are clearly evident in Figures 1 and 2. They indicate a 13 and 10 per cent increase in flow adequacy at the KE and KW reactors, respectively.

b. Crosstie Line Corrosion Inhibition

It is apparent that something needs to be done to reduce the corrosion rate in these crosstie lines. Either corrosion inhibition, such as painting, etc., must be done or the maintenance problem accepted of line cleaning every one to two years. Reducing the corrosion, by means of painting or cement coating will probably cause some small decrease in flow below that of the cleaned pipe, but should still yield at least a 7 per cent increase in emergency flow above the conditions expected this summer.

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c. Design Change #772

The completion of this change is expected to improve the electrical reliability of the secondary system,⁽⁹⁾ to increase the number of pumps at the aiding reactor, and to increase the adequacy of the last-ditch cooling.

d. Flow Measurement Accuracy

The reasons for the considerable scatter of the crosstie flow measurements is not known. Therefore, it will be difficult to reduce this factor. A contributing condition is the variation in diesel pump speed. This may vary as much as 2 per cent with a significant effect upon the emergency flow. At the present time, no allowance for improved flow accuracy is included in this survey.

e. Crosstie Coolant Temperature

Somewhat arbitrarily the crosstie coolant temperature was assumed to be 5 C above the process water temperature. It is probable that an increased crosstie bleed combined with a measurement of this temperature in each of the K-Reactor valve pits would yield a maximum temperature increase above process water of only 2 to 3 C. Thus, another 2 to 3 per cent potential increase in cooling adequacy exists.

f. Increased Diesel Pump Speed

A test was performed on these pumps, and it is expected that their speed will be increased from 720 to 750 rpm by June 1. This is expected to yield a 4 per cent flow increase. This corresponds to a 3 per cent increase in cooling adequacy when considering the effect of the increased reactor heat extraction requirements at the shorter time of this increased flow takeover from the flywheel decay. This increase in adequacy has been included in the last-ditch cooling adequacy curves presented as Figures 1 and 2 of this report.

CONCLUSIONS

This adequacy analysis of the Production Test results indicates a continued need for concern about this problem. The completion of the following relatively minor system improvements are recommended to increase the cooling reliability and adequacy of this last-ditch cooling system.

1. Clean and coat the high pressure crosstie lines.
2. Monitor the crosstie coolant temperature and flush to maintain within 2°C of process.

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3. Complete Design Change #772 to obtain an improvement in the reliability and number of the secondary system pumps that would remain running under emergency conditions.
4. Increase the diesel pump speed from 720 to 750 rpm, and test to determine speed reliability.

After the completion of these relatively minor and inexpensive improvements in the K last-ditch system, we may be able to relegate this problem from one of continued concern and expensive testing to that of routine annual monitoring of the flow characteristics of the crosstie. We should have a 10 per cent excess cooling capacity and need no longer consider this a current safety problem.

ACKNOWLEDGMENT

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APPENDIX A

NOMENCLATURE

- F_C = Fraction of clean line flow to stricken reactor for corrosion flow-loss effect.
- F'_C = Fraction of clean line pressure-drop for corrosion resistance build-up.
- F_D = Fraction of diesel pumping flow to stricken reactor.
- F_ϵ = Flattening efficiency, ratio of bulk temperature rise to the temperature rise in ten hottest standard tubes.
- $1+F_G$ = Low flow error factor, fractional increase in measured flow to obtain assured flow.
- F_S = Fraction of measured flow that can be counted on during an emergency because of possible line 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_D = Total diesel pumping flow - 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 - °C.
- T_{ic} = Crosstie water inlet temperature - °C.
- X = Years since line cleaning.

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NOMENCLATURE (Continued)

- α = Lumped flywheel decay parameter.
- τ = Time after reactor safety rod scram - seconds.
- p_D = Diesel pump discharge pressure - psig.
- p_J = Pressure at junction of diesel pump discharge line and the crosstie - psig.
- p_R = Riser pressure at bottom (BORP) - psig.
- Δp_{fD} = Pressure-drop (friction) diesel pumps to junction - psi.
- Δp_{fC} = Pressure-drop (friction) junction to risers (BORP) - psi.
- Δp_{fR} = Reactor frictional pressure-drop - psi.
- Δp_{fT} = Total frictional pressure-drop junction to downcomer - psi.

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APPENDIX B

DIESEL PUMP AND CROSSTIE JUNCTION PRESSURES

1. Diesel Pump Characteristics

The results of a number of tests^(3, 4, 5) have been summarized to obtain the best value of diesel pump flow characteristics. The experimental results are tabulated in Table A-I, and the diesel pump characteristic curve is shown in Figure A-1. There is considerable scatter in the test results and any effect of which two of the three pumps are operating is believed to be well within the over-all variation of the test data.

These results are for a diesel pump speed of 720 rpm. This is the present normal diesel pump operating speed for this system. An increase in pump speed to 750 rpm is expected, and theoretical 750 rpm pump curves are presented.

2. Crosstie Line Junction Pressure

The diesel pumps provide flow to the crosstie system through a 36-inch diameter manifold line about 250 feet long. The full diesel pump flow goes through this line and splits at these crosstie junctions (two crosstie parallel lines) to go to either the KE or the KW reactor. To determine the line pressure drop, it is necessary to determine this junction pressure as a function of the total diesel pump flow and then the line pressure drop as a function of the flow to each reactor. The crosstie junction pressures are also tabulated in Table A-I and the line pressure drop from the pump discharge to the junction is given by the equation*

$$\Delta p_{f_D} = p_D - p_J + 9 \quad (A-1)$$

It is significant to note that the reported junction pressures in Reference 3 gave line pressure drops of -1.5 to +2.7 psi compared with line pressure drops in the latter tests^(4, 5) of 3 to 7.5 psi. Therefore, it was arbitrarily assumed that the junction pressures in Reference 1 were 5 psi too high. In addition to the fact that this makes the results of the three tests comparable, it is obvious that the negative pressure-drop values in the Reference 3 tests are in error. The resultant pump manifold pressure drop with flow is shown in Figure A-2.

*Where the value of 9 psi represents the static pressure increase between the pump discharge and the crosstie junction.

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TABLE A-I

EXPERIMENTAL HPCT JUNCTION PRESSURE
AND Δp_f DIESEL PUMPS TO HPCT (720 rpm)

Test No.	Pump No.	Diesel Pumps			Junction Pressure		$p_D - p_J$ psi	Frictional Pres. Drop Δp_{fD} psi
		No. of Oper. Pumps	Flow 1000 gpm G_D	Discharge Pressure p_D	p_J psi	p_J^* psi		
PT-IP-499								
5.4.5-d	1,2	2	37.0	76.5	86.0	81.0	-4.5	4.5
f	1,2	2	32.0	88.5	99.0	94.0	-5.5	3.5
f-1	1,2	2	36.0	78.5	89.0	84.0	-5.5	3.5
g	1,2	2	38.7	70.5	76.8	71.8	-1.3	7.7
h	All	3	47.5	83.0	92.0	87.0	-4.0	5.0
i	All	3	50.1	80.7	88.0	83.0	-2.3	6.7
j-1	All	3	53.8	78.0	85.0	80.0	-2.0	7.0
k	2,3	2	43.2	58.5	66.0	61.0	-2.5	6.5
l	2,3	2	40.2	67.5	76.0	71.0	-3.5	5.5
n	All	3	46.9	89.5	98.0	93.0	-3.5	5.5
q	2,3	2	35.4	82.0	91.0	86.0	-4.0	5.0
s	2,3	2	33.4	88.5	97.0	92.0	-3.5	5.5
5.4.6-b	1,2	2	34.1	85.0	94.0	89.0	-4.0	5.0
c	1,2	2	37.1	76.0	85.0	80.0	-4.0	5.0
g	1,2	2	38.6	73.5	81.0	76.0	-2.5	6.5
h	All	3	48.9	88.0	94.0	89.0	-1.0	8.0
i	All	3	51.1	84.0	91.0	86.0	-2.0	7.0
h-1	All	3	46.6	90.0	98.0	93.0	-3.0	6.0
PT-IP-573								
B- 1	1,3	2	40.2	74.0	77.5	-----	-3.5	5.5
B- 8	All	3	54.8	84.0	85.5	-----	-1.5	7.5
B-11	1,3	2	35.9	88.0	92.0	-----	-4.0	5.0
B-14	1,3	2	40.5	76.0	79.5	-----	-3.5	5.5
B-14'	2,3	2	40.1	75.0	79.5	-----	-4.5	4.5
PT-IP-628								
A- 1	1,2	2	37.5	73.5	79.5	-----	-6.0	3.0
A- 3'	1,2	2	37.7	75.0	77.5	-----	-2.5	6.5
A- 2	1,2	2	39.7	67.5	69.5	-----	-2.0	7.0
A- 3	1,2	2	37.8	73.5	77.5	-----	-4.0	5.0
A- 4	1,2	2	37.0	77.5	80.5	-----	-3.0	6.0

$$\Delta p_f = p_D - p_J + 9$$

*Corrected junction pressure for assumed +5 psi error

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NUMBER OF
*PUMPS

750 rpm

750 rpm

15	20	25	30
Total Diesel Pump, Flow, 1000 gpm			

FIGURE A-1

LOOKED AT
PUMP DISCHARGE PRESSURE

٢٠

ECT JUNCTION SYSTEM

scharge

Junction Pressures

50

1797

1951

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(3) The Commission (1925-26)

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1000

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Total Diesel Pump Flow, 1000 gpm

*Testing Date	
W. H. Hedrick (11-1-53)* (10)	2.3 X
W. H. Hedrick (11-1-53)* (10)	1.2 X
L. C. Tessor (2-1-53)* (5)	1.1 X
W. H. Hedrick (11-1-53)* (10)	1.3 X
D. L. Remington (11-1-53)* (10)	2.3 X
W. H. Hedrick (11-1-53)* (10)	1.2 X
W. H. Hedrick (11-1-53)* (10)	1.1 X

FIGURE A-2
LINE PRESSURE DROP
DIESEL PUMPS TO HPCT JUNCTION

APPENDIX C

CROSSTIE LINE CORROSION EFFECTS

1. Crosstie Line Frictional Pressure Drop

This was determined for the three testing programs in the following manner. For every case where the crosstie flow was the sole source of reactor cooling the crosstie flow, as indicated by the flow tube, and the reactor A and B riser pressure were noted. The corresponding junction pressure was obtained for the given flow and diesel pumping conditions from Figure A-1. This averages out the experimental manifold pressure-drop data and assumes no effects of corrosion on this value. Considering the low pressure drop and the large size of the manifold line this assumption involves negligible error in our results. With the crosstie flow and the pressure at each end of the line determined, the frictional pressure drop was readily determined from the equation

$$\Delta p_{f_c} = p_j - p_B - 3.5 \quad (A-2)$$

where the 3.5 represents the static pressure elevation of the BORP pressure above the crosstie junction. This determination is tabulated in Table A-II and shown graphically in Figure A-3.

2. Corrosion Effect

The increase in crosstie line pressure drop with corrosion is indicated by an increase in the value of the ordinates in the curves in Figure A-3. For instance, there was a significant decrease in the value of the pressure drop in Figure A-3 between the first and second tests. All but the new parallel portion of the crosstie had been in service for over 8 years and about a one-half inch coating of very rough corrosion product was observed on components of the old piping at the time of the first testing series.

Just before conducting the second testing series, the lines were mechanically cleaned. As indicated in Figure A-3, the average effect of this line cleaning was to reduce the frictional pressure drop by 50 per cent. Ten months later the third test series was conducted. The results of this test, also shown in Figure A-3, indicated renewed corrosion and increased line pressure drop for a given crosstie flow.

The pressure-drop data in these tests showed a significant experimental scatter. Therefore, to obtain the best results, a frictional pressure-drop curve with a 1.85 slope was drawn so as to average the results of the high flow tests in each testing series. Although all test flows were well into the turbulent flow region, which has a 1.85 slope, the

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APPENDIX C (Cont.)

FIGURE A-3.
CROSS-TIE LINE FRACTURE DROP
(15 C Temperature)

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flow accuracy was considered greater at the higher flows. The frictional pressure drop follows the basic relationship

$$\Delta p_{f_c} = k G_c^{1.85} \quad (A-3)$$

With this crosstie pressure drop determined, our goal was to obtain a corrosion flow factor. The fractional increase in total friction pressure drop between the pumps and the reactor downcomer was determined from the relationship

$$F_c' = \left[\frac{\Delta p_{f_c} x + \Delta p_{f_R}}{\Delta p_{f_c 0} + \Delta p_{f_R}} \right] G \quad (A-4)$$

where

F_c' = Fractional increase in total pressure drop at constant flow between the clean line and the line after x years of corrosion.

Δp_{f_c} = The crosstie pressure drop psi

and Δp_{f_R} = The reactor pressure drop psi shown in Figure 32 of Reference 11.

In this equation, the subscript 0 refers to clean line flow. There is negligible corrosion effect in the reactor, therefore Δp_{f_R} is constant and all three pressure-drop values are for the same crosstie and reactor flow. Using this equation values of F_c' were determined as shown in Table A-III for the three test cases of cleaned lines, and 0.7, and 8 years of corrosion.

The corresponding flow decrease at constant available pressure drop for this corrosion effect is given by the relationship:

$$F_c = \left[\frac{1}{F_c'} \right]^{.54} \quad (A-5)$$

where

F_c = the fraction of clean line flow

The foregoing procedure is an approximation that appears justified by the inaccuracies apparent in the experimental data. The rigorous method involves a calculation of the diesel pump supply pressure as delivered at the reactor A and B risers, and matching this at the same flow conditions with the reactor flow-pressure curve. This requires a double trial-and-error solution for which there is neither time nor justification. Values of F_c , determined in the aforementioned approximate manner, are presented in Table A-III and in Figure 3 in the body of the report.

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TABLE A-II

EXPERIMENTAL HPCT LINE - Δp_f

Test	Diesel Pumps			Reactor			Crosstie					
	No.	Flow	Junct. Pres.	No.	Cond.	Riser Pres. P_{RA}	HPCT		Date		Flow Fract. F_D^{**}	
							Temp.	Flow G_C	Δp_{fc}	Temp.		Δp_f
PT-IP-499 5.4.5-d	2	37.0	83	KE	Aiding	72	17°	8.8	7.5	7-21-62		0.76
				KW	Stricken	41		28.2	38.5	15°	*	
f	2	32.0	94	KE	Aiding	45		32.0	45.5			1.00
f-1	2	36.0	85	KE	Aiding	76		7.0	5.5			
				KW	Stricken	42		29.0	39.5			0.81
g	2	38.7	78	KE	Aiding	62		12.2	12.5			
				KW	Stricken	32		26.5	42.5			0.68
h	3	47.5	91	KE	Aiding	66		17.6	21.5			
				KW	Stricken	45		29.9	42.5			0.63
i	3	50.1	86	KE	Aiding	54		21.4	28.5			
				KW	Stricken	41		28.7	42.0			0.57
k	2	43.2	62	KE	Stricken	38		21.0	20.5			
				KW	Aiding	40		22.2	18.5			0.51
l	2	40.2	73	KE	Aiding	55		14.7	14.5			
				KW	Stricken	38		25.5	31.5			0.63
h	3	46.9	92	KE	Aiding	76		15.6	12.5			
				KW	Stricken	45		31.3	43.5			0.67
q	2	35.4	87	KE	Aiding	81		5.5	2.5			
				KW	Stricken	43		29.9	40.5			0.85
s	2	33.4	91	KE	Aiding	77		15.2	10.5			
				KW	Stricken	76		18.2	11.5			0.54
5.4.6-b	2	34.1	90	KE	Stricken	76		18.7	10.5			0.55
				KW	Aiding	67		15.4	19.5			
c	2	37.1	82	KE	Stricken	39		27.6	39.5			0.72
				KW	Aiding	72		10.3	6.5			
g	2	38.6	78	KE	Stricken	37		25.4	37.5			0.66
				KW	Aiding	65		13.2	9.5			
h	3	48.9	88	KE	Stricken	41		28.9	43.5			0.59
				KW	Aiding	69		20.0	15.5			
i	3	51.1	84	KE	Stricken	41		27.6	39.5			0.54
				KW	Aiding	62		23.5	18.5			
h-1	3	46.6	92	KE	Stricken	42		29.4	46.5			0.63
				KW	Aiding	78		17.2	10.5			

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TABLE A-II (Continued)

Test	Diesel Pumps			Reactor			Crosstie					
				No.	Cond.	Riser Pres. P_{RA}	HPCT		Date		Flow Fract. F_D^{**}	
	No.	Flow	Junct. Pres.				Temp.	Flow G_C	Δp_{fc}	Temp.		Δp_f
PT-IP-573											2-16-63	
B-1	2	40.2	74	KE	Stricken	43	50	30.9	27.5	150	25.4	0.77
				KW	Aiding	72		9.3	-1.5		----	
B-8	3	54.8	80	KE	Aiding	68		21.0	8.5		7.8	
				KW	Stricken	48		33.8	28.5		26.3	0.62
B-11	2	35.9	86	KE	Stricken	54		0	----			
				KW	Stricken	51		35.9	31.5		29.1	1.00
B-14	2	40.5	74	KE	Aiding	74		8.9	-3.5		----	
				KW	Stricken	46		31.6	24.5		22.6	0.78
B-14'	2	40.1	74	KE	Aiding	74		8.6	-3.5		----	
				KW	Stricken	46		31.5	24.5		22.6	0.78
PT-IP-628											11-1-63	
B-1	2	40.2	74	KE	Stricken	43	140	30.9	27.5	150	*	0.77
				KW	Aiding	74		9.3	-3.5			
A-1	2	37.5	82	KE	Stricken	41		28.9	37.5		*	0.77
				KW	Aiding	71		8.6	7.5			
B-14'	2	40.1	74	KE	Aiding	74		8.6	-3.5			
				KW	Stricken	46		31.5	24.5			0.78
A-3	2	37.7	82	KE	Aiding	72		7.8	6.5			
				KW	Stricken	44		29.9	34.5			0.79

*The Δp_f values at 15°C are assumed equal to the experimental values at 17°C

**The fraction of the diesel pump flow going to the stricken reactor

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TABLE A-III
CROSSTIE CORROSION FLOW FACTOR

Reactor	Crosstie Flow 1000 gpm	Corrosion		Reactor Pressure Drop ΔP_{IR}	Total ΔP ΔP_{IT}	Corrosion Factors	
		Years	Pressure Drop ΔP_{IC}			F_C Pres.	F_C Flow
KE	10	0	3.3	3	6.3	1.00	1.00
		0.7	4.5		7.5	1.20	0.90
		8.0	7.0		10.0	1.60	0.77
	20	0	12.0	9	21.0	1.00	1.00
		0.7	16.0		25.0	1.20	0.90
		8.0	25.0		34.0	1.60	0.77
KV	30	0	25.0	20	45.0	1.00	1.00
		0.7	34.0		54.0	1.20	0.90
		8.0	53.0		73.0	1.60	0.77
	10	0	2.7	3	5.7	1.00	1.00
		0.7	3.5		6.5	1.10	0.95
		8.0	5.0		8.0	1.40	0.83
	20	0	10.0	9	19.0	1.00	1.00
		0.7	12.5		21.5	1.00	0.95
		8.0	18.0		27.0	1.40	0.83
	30	0	20.0	20	40.0	1.00	1.00
		0.7	27.0		47.0	1.20	0.90
		8.0	38.0		58.0	1.45	0.83

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