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K-REACTOR WATER PLANT ANALYSIS

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December 22, 1958

W. K. MacCready
Acting General Manager
IRRADIATION PROCESSING DEPARTMENT

K-REACTOR WATER PLANT ANALYSIS

The reliability of the K-Reactor water plants has been reviewed by Research and Engineering personnel; this study augments a brief analysis of the KW generator failures incident given in my letter of October 6, 1958. Our attention has been directed largely to ascertaining the requirements of the system, the consequences of possible component failures, giving general assessment of the primary features upon which reliability depends and identifying potential improvements which would increase reliability or moderate the consequences of a major failure. Our attention is restricted to failures which might occur as a result of equipment failure, operator error or certain natural causes such as earthquakes; we have not weighed special considerations brought on by enemy action such as sabotage, for example, in this evaluation. R. S. Bell is providing an analysis of the operating and maintenance aspects of K water plant reliability to complement this engineering study.

Summary and Conclusions

It is concluded that the basic elements of the system lend themselves to reliable operation. The lines of defense appear adequate in depth, that is, a primary system operated with BPA power, an emergency "backup" system consisting of two essentially independent steam driven generators driving selected pumps, and a "last ditch" system providing coolant from an independent area through hydraulic cross-tie lines. It is further concluded that the reliability of the system currently is good and is very intimately associated with level of the maintenance and testing programs and competence with which they are carried out. The K-Reactor water plant systems are complex and can be quite vulnerable to both equipment failure and operator error; however, we conclude that substantial progress has been made in these important areas.

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There remains some question relative to the consequences of several possible failures in the plant; answers to these largely depend upon definition of the conditions under which the high lift pumps will cavitate and the flow capability under cavitating conditions. Experiments in K Plant will be required to resolve these points which have an impact on the power levels and flow rates which can be prudently attained. Permissible process conditions are involved rather than inherent plant safety in the sense that failure probabilities are not influenced.

There are improvements which can be made to the plant but the justification depends upon a quantitative assessment of the increment of safety obtained; for the most part this picture will not be clarified until the pump cavitation studies are performed. However the installation of a 4.16 KV electrical cross-tie has been recommended previously as justified and its installation should proceed.

Discussion

A) K-Reactor Water Plant

Simplified schematic diagrams of the dual K-Area water supply systems are shown in Figures 1 and 2 of Appendix I. A simplified schematic diagram of the electrical power supply system is shown in Figure 3 of Appendix I.

The K-Reactor flow requirements are provided by six pump sets at each reactor. A pump set consists of two pumps, a low lift pump (100 ft. TDH at 28,000 gpm) supplying the suction requirements of the high pressure pumps (870 ft. TDH at 28,000 gpm). Both the low and high lift pumps are electrically driven: the low lift pumps by 4.16 KV BPA power with two independent steam driven generators providing alternate power sources in each area and the high lift pumps by 13.8 KV BPA power with an electrical intertie between Plants providing a backup power source in event of local BPA power interruption at one K plant. Low and high pressure hydraulic cross-tie lines also interconnect the two K plants - the low pressure cross-tie system supplied by the service pumps and the high pressure cross-tie by the high lift pumps combined with the low lift pumps, or by the low lift pumps alone when the companion reactor is down.

The normal flow requirements are then supplied through the primary power source - BPA. In the event of a complete BPA power failure, the reactor is shut down automatically and the shutdown flow is supplied by pump flywheel energy during the first minute and by the emergency steam driven generators, coupled to selected low lift pumps, thereafter; these generators are now electrically independent and either is capable of supplying the full shutdown flow requirement to its own reactor plant. The steam driven generators then constitute the "backup" system although they drive the same low lift pumps which are driven from BPA power in normal operation; in this sense the primary and backup systems are not independent sources

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of cooling although this dual use of common components is not assessed to be a significant weakness of the system.

The "last ditch" cooling is provided from the companion area via a low pressure cross-tie line (from service water pumps) and a high pressure cross-tie line (from the low and high lift process water pumps). The alternate sources of coolant and/or power are summarized in the following table:

TABLE I

<u>System</u>	<u>Source of Power</u>	<u>No. of Pumps</u>	<u>Flow provided*</u> (gpm x 10 ⁻³)
1) <u>Primary</u> (High-Lift & low-lift pumps in series)	BPA	6 sets 5 sets 4 sets	177 163 90-135
2) <u>Backup</u>	Emergency Generators	6 5 4	59 56 53
(One generator only)		3 2 1	50 42 30
3) <u>Last Ditch</u>	Companion Reactor BPA and/or Emergency Generators		
Both Cross-Ties** (Other area operating)			31
Both Cross-Ties ** (Other area down)			19
Low Pressure Cross-Tie Only			9.5

* Flow provided to the reactor by the itemized element of the system assuming no pipe ruptures, reactor loaded with I & E slugs and pumps operating out of the region of cavitation, except where double values are given. The lower value then indicates estimated flow resulting from possible severe cavitation at reduced back pressure.

** When the other area is operating the high pressure cross-tie is supplied with pressure by the low-lift and high-lift pumps operating in series. When the other area is down the high pressure cross-tie is supplied with reduced pressure by the low-lift pumps only. The low pressure cross-tie is always supplied by a separate set of service water pumps in each area.

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The flow requirements during operating and shutdown conditions during the summer (winter conditions require less flow) are shown in Figure 4 of the Appendix I on the basis of 1) requiring sufficient pressure to sweep steam from the fuel channels should it be formed and 2) requiring sufficient flow to prevent an increase in bulk temperature above 90°C. Shown also are the flow capabilities of ranging combinations of the "backup" and "last ditch" systems when complete failure of BPA power is assumed. The following features are pertinent:

- 1) In the period from 0-10 seconds the flow provided by pump flywheel energy is not sufficient to preclude a bulk coolant temperature rise. This condition is similar to that in all reactors and typifies a situation which results in elevating bulk effluent temperatures. The temperature surge in this interval must be kept within limits as necessary to avoid effluent system damage which would result from excessive steam formation.
- 2) The combination of the flow provided by flywheel decay and one of the two emergency generator driven combinations of low-lift pumps in the stricken area is designed to provide sufficient flow so that bulk coolant temperatures do not exceed 90°C.
- 3) The low pressure cross-tie line alone will provide adequate cooling after about one hour.
- 4) There are cases not shown in which the high lift pumps may go into cavitation. This could result from loss of the low lift system (4.16 KV power), as in the August 21, 1955 KW emergency generator failure incident in which the high lift pumps cavitated. Cavitation possibly could result from a pump shaft break on a pump in either set. This contingency is protected against by maintenance and inspection programs and in establishing process requirements. However, more data are required to define the risk from these sources at elevated flow rates and power levels.

B) Assessment of System Reliability

The reliability of the K water plants cannot be expressed quantitatively any more than can be done for any electrical or mechanical system. It is possible to derive a statistical approach in expressing the reliability of some elements; in the case of others the failure frequency is too low (BPA power for example) and in these cases one is prone to assume reliability. The statistical approach can also be misleading because the consequences of failure are potentially very large; if a statistical approach is accepted as a basis for judging adequate reliability, rather than relative reliability, one also accepts an expectation of loss and to do this it is necessary to place a value on an incident of given magnitude and its consequences. Unfortunately, there are many consequences of a serious reactor incident which are difficult to assess quantitatively and these may out-rank the obvious economic consequences in a scale of values held by a responsible organization. Therefore, the use of statistics in assessing reliability of the K water plant is in a relative and qualitative sense.

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1) Primary Cooling System

The continuity of flow from the primary cooling system depend upon continuity of BPA power which has proven quite reliable. There have been no instances in the history of Hanford operations of BPA power lost at the Midway substation without prior warning; in only one case, June, 1950, has all BPA power been lost - this with warning - and no damage resulted. On one occasion all BPA power was lost to the D-DR dual reactor area without warning - this as a result of a substation operator switching error - and the steam backup system operated as designed. There have been twelve instances of BPA power disturbances during the period 1951-1958 which caused reactor shutdowns; these disturbances were characterized as voltage or frequency fluctuations which scrambled reactors without other consequences. There have been seven instances of line failure in one leg of the BPA loop which required reactor shutdowns because only a single source of BPA power remained; five of these resulted from pole fires, one was caused by a construction crane and one resulted from a guy wire anchor testing accident. While this reliable performance serves to provide considerable comfort we do not deem the BPA power supply sufficiently reliable, in itself, to provide a justification for any but the highest order of protection from both a "backup" and "last ditch" system.

The piping in the primary system appears conservatively designed for the low pressure and temperature service required (about 650 psi at water temperatures ranging below 25°C). The earthquake protection also appears adequate for this region; however this point will be being further explored in studies now being initiated.

The consequences of arbitrarily selected complete line failures are quantitatively expressed in Table II. It is seen that instantaneous failure of a riser could be experienced (FB-5 valve closed) without major reactor damage in the case of six pump operation if pump cavitation with associated severe flow decreases do not accompany the incident; the degree of conservatism in the cavitation assumption is not now known but will be determined experimentally.

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TABLE II
K REACTOR LINE RUPTURE EFFECTS

<u>Charge</u>	<u>Rupture Condition</u>	<u>No. Pumps</u>	<u>Estimated Damage</u>
Solid	Rupture in High	6	None
	Pressure Cross-tie	6	None*
	Line	5	None
		5	Some Central Zone Melting*
I & E	Rupture in High	6	None
	Pressure Cross-tie	6	Effluent System Damage*
	Line	5	Effluent System Damage
		5	Some Central Zone Melting*
Solid	Rupture in A or B	6	Some Central Zone Melting <u>Catastrophic*</u>
	Riser		
	FB-5 Valve open		
Solid	FB-5 Valve closed	6	Effluent System Damage Some Central Zone Melting*
Solid	Rupture in A or B	6	None
	Riser		
	FB-5 closed, A and	5	Some Central Zone Melting*
	B risers separated		
		5	Effluent System Damage Some Central Zone Melting*
I & E	Rupture in A or B	6	Effluent System Damage Some Central Zone Melting*
	Riser		
	FB-5 Closed, A and	5	Effluent System Damage <u>Catastrophic*</u>
	B risers separated		

*These values represent the possible effect of assumed severe pump flow reduction due to cavitation.

The consequences of a single pump failure in the primary system, as could be typified by a "shaft break" are shown in Table III. Some reactor and effluent system damage could result; the degree again depending upon the cavitation assumptions.

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TABLE III
K-REACTOR PUMP LOSS EFFECTS (Shaft Break)

<u>Event</u>	<u>Number of Pumps Operating**</u>	<u>Operating Conditions</u>	<u>Effluent System Damage</u>	<u>Reactor Damage</u>
One pump failure		1) Both reactors at full level		
Solid Charge 160,000 gpm	(6-pump Operation)	2) Both with FB-5 cross-tie valves open	None	None
One pump failure	(6-pump Operation)	1) Both reactors at full level	None	None
I&E Charge 177,000 gpm		2) Both with FB-5 cross-tie valves open		
One pump failure	(6-pump Operation)	1) Both reactors at full level	Minor-Moderate*	None-Minor*
I&E charge, Zr Tubes 190,000 gpm		2) Both with FB-5 cross-tie valves open		
One pump failure	(5-pump Operation)	1) Both reactors at full level	Minor-Moderate*	None-Minor*
Solid charge 155,000 gpm		2) Both with FB-5 cross-tie valves open		
One pump failure	(5-pump Operation)	1) Both reactors at full level	Minor-Severe*	Minor-Severe*
I&E charge 163,000 gpm		2) Both with FB-5 cross-tie valves open		
One pump failure	(5-pump Operation)	1) Both reactors at full level	Minor-Severe*	Minor-Severe*
I&E charge, Zr tubes 162,000		2) Both with FB-5 cross-tie valves open		

* The lesser amount of damage would occur if these pumps maintain maximum flow under severe cavitation conditions for loss of back-pressure.

** Six pump operation is expected to be the most frequent mode.

2. Backup Cooling System

The "backup" cooling systems, the two independent steam driven emergency generators each coupled to selected low lift pumps, has been called upon to function once and failed. A frequency surge in the BPA system resulted in a 4.16 KV power loss at KW Reactor. The emergency generators failed to pick up the load and the low lift pumps were without power. The high lift pumps, which are driven with 13.8 KV power, continued to operate but cavitated severely when the flow from the low lift pumps decayed. The pressure dropped rapidly to the point where additional flow was provided from KE Reactor through the high pressure cross-tie line whereupon the cavitating high lift pumps were removed from service.

Either of the two emergency generators has the capacity to supply the required shutdown cooling independently (see Figure 4 of Appendix I); the failure in the above August 21, 1955 incident was the result of faulty relay adjustment and maintenance and inadequate generator maintenance and testing. Basic modifications have been made in the electrical circuitry to isolate the generators electrically from each other and an aggressive program of maintenance and testing undertaken. The response of the emergency generators to simulated BPA failures has now been tested over three hundred times without failure; this indicates a statistical reliability of 99 percent at the 95 percent confidence level. An indication of the sensitivity of the relative "expectation of loss" to the number of independent generators as well as the reliability of operations is discussed in document HW-54644, "K Primary Pump Power Cross Feed", K. W. Hess and is shown in Table IV.

TABLE IV
TEN YEAR EXPECTATION OF LOSS

<u>PARAMETER</u>	<u>BPA FAILURE FREQUENCY</u>	<u>NO. OF EMERGENCY GENERATORS</u>	<u>EMERGENCY GENERATOR RELIABILITY</u>	<u>DISASTER COST</u>	<u>TEN YEAR EXPECTATION OF LOSS</u>
Effect of BPA Failure	1/10 yrs.	2	95%	$\$2 \times 10^9$	\$ 5,000,000
Frequency	1/20 yrs.	2	95%	$\$2 \times 10^9$	\$ 2,500,000
	1/50 yrs.	2	95%	$\$2 \times 10^9$	\$ 1,000,000
	1/100 yrs.	2	95%	$\$2 \times 10^9$	\$ 500,000
Effect of Generator Reliability	1/10 yrs.	2	95%	$\$2 \times 10^9$	\$ 5,000,000
	1/10 yrs.	2	90%	$\$2 \times 10^9$	\$20,000,000
	1/10 yrs.	2	99%	$\$2 \times 10^9$	\$ 200,000
	1/10 yrs.	2	99.9%	$\$2 \times 10^9$	\$ 2,000
Effect of Disaster Cost	1/10 yrs.	2	95%	$\$2 \times 10^9$	\$ 5,000,000
	1/10 yrs.	2	95%	$\$500 \times 10^6$	\$ 1,250,000
	1/10 yrs.	2	95%	$\$1 \times 10^9$	\$ 2,500,000
	1/10 yrs.	2	95%	$\$5 \times 10^9$	\$12,500,000
Effect of Generator Redundancy	1/10 yrs.	2	95%	$\$2 \times 10^9$	\$ 5,000,000
	1/10 yrs.	3	95%	$\$2 \times 10^9$	\$ 250,000
	1/10 yrs.	4	95%	$\$2 \times 10^9$	\$ 12,500
Effect of Generator Redundancy	1/10 yrs.	2	99%	$\$2 \times 10^9$	\$ 200,000
	1/10 yrs.	3	99%	$\$2 \times 10^9$	\$ 2,000
	1/10 yrs.	4	99%	$\$2 \times 10^9$	\$ 20

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The above indicates marked improvement in increased backup system reliability (reduced expectation of loss) as the generator reliability is increased and also as the number of independent generators are increased. A 4.16 KV electrical cross-tie between the emergency generator sets in the two K areas could approach the effect of doubling the number of generator sets available in the backup system at a given reactor and has been recommended.

Basically, however, we assess the backup system to be as strong as the maintenance and reliability testing programs maintain it; the recent generator testing statistics would indicate good reliability.

3. "Last Ditch" Cooling System

The "last ditch" cooling system consists of a low pressure hydraulic cross-tie line (fed by service water pumps) and a high pressure hydraulic cross-tie line (fed by both the high lift and low lift pump sets) which interconnect the KE and KW Reactor cooling systems to provide an emergency cooling source from an independent reactor plant. Operating difficulties with elements of this system, particularly the V-73 valve which must function to admit coolant to a stricken reactor, have been experienced but emphasis on maintaining the system has increased its reliability. The system has the strong feature that it can provide an adequate source of shutdown cooling to a stricken reactor indefinitely following an interim period ranging to 15 minutes after shutdown during which the capacity may be inadequate to prevent boiling and some reactor damage at the higher range of predicted K Reactor power levels. This situation has been analyzed and we believe it is not necessary to restrict the K Reactor power levels to a limit calculated to prevent boiling in a situation where BPA power and both emergency generators are lost concurrently as long as emergency generator "reliability" is maintained at a high level. There are alternate methods of compensating for this "growth beyond the cross-tie capacity" which are being considered and will be recommended if the increment of added safety gained justifies the expenditure.

C) Methods of Improving Reliability

Research and Engineering believes that maintenance and testing programs are the factors which determine the reliability of the system now in place. Progress has been made in these areas as well as in improving the plant -- significant improvements include:

1. The emergency steam-electric generators have been provided with all-steam auxiliaries to make the system independent of electrical power.
2. The generators are operated in a nearly independent fashion, instead of in parallel as in original operation.
3. The training of personnel has been emphasized. Maintenance has been emphasized.
4. Routine testing of the ability of a single generator set to assume its emergency load has been established.
5. The operation of the V-73 valve in the cross-tie line is routinely tested.

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There are further plant modifications which would improve the reliability or decrease the consequences of a failure of a major element in the system. These potentially include:

1. Provide a seventh pump set for each reactor; this would permit continuous balanced operation with the FB-5 valve closed with attendant reduction in damage from riser failure and decrease the consequences of a "pump shaft break" experience.
2. Provide check valves in the cross-tie import to each of the A and B risers. This again would moderate the effect of a riser break.
3. Increase the capacity of the "last ditch" system under circumstances where the companion reactor is shutdown.
4. Provide a 4.16 KV electrical cross-tie between the emergency generators in the companion plants.

Alternates 1, 2, and 3 require more study before they can be definitely recommended; in particular much more needs to be known about high lift pump cavitation and the resulting flows under "accident" conditions. Alternate 4 appears to be justified and has been recommended.

RL Dickeman

Acting Manager
Research & Engineering
IRRADIATION PROCESSING DEPARTMENT

RL Dickeman/bs

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APPENDIX I

DESCRIPTION OF THE K-WATER PLANTS AND THEIR CAPABILITIES

The major features of a single K-Reactor coolant supply system are shown in Figure 1. Coolant water is pumped from the river to the head-house where water treatment is initiated. After a settling period the water is filtered and stores in the clear-wells which have a total capacity, for each reactor, of 18×10^6 gallons. All water for reactor and service use is normally drawn from these clear-wells. The high and low pressure cross-ties provide emergency coolant to or from the companion reactor. The high pressure cross-tie connects the reactors at a cross-tie line joining the A and B risers. Process water is supplied to the reactor by low-lift and high-lift pumps operating in series. Three pump sets serve each side of the reactor. The systems serving each side of the reactor are partially connected by the FB-5 valve.

Water plant connections between the two K-Plants are shown in Figure 2. Pressure in the low-pressure cross-tie is maintained against check valves held closed by the higher process line pressure at pumps 3 and 4. Upon loss of process line pressure these check valves open to provide service pump driven flow from either area. In the high pressure cross-tie, pressure is maintained by each area against the V-73 valve adjacent to and controlled by the companion area. The V-73 is an automatic, air actuated, quick acting valve controlled by the adjacent area. Opening this valve permits flow into the A and B risers. Flow from the A and B risers is automatic upon demand by the companion area.

The BPA and emergency generator electrical power distribution system pertinent to the primary and backup coolant supply systems of the dual K-Reactors is shown in Figure 3 in a simplified single-line schematic diagram. It is seen that three sources of 230 KV power supply two transformers in each area. The resulting two sources of 13.8 KV power in each area are supplemented by a 13.8 KV cross-tie between the areas. The 13.8 KV system provides power for the six high-lift pumps in each area, and for four transformers in each area which yield power at 4.16 KV. The 4.16 KV system powers the six low-lift pumps in each area, the service water pumps, the river pumps and other service lines. The two 4.16 KV transformers in each area which power the low-lift pumps are backed up by three 5000 KVA steam powered 4.16 KV generators, two of which are always on the line. In the event of loss of 4.16 KV power from the transformers the two generators automatically assume the load, which is reduced by elimination of most of the river pumps.

The flow requirements to maintain adequate cooling and to provide "steam-sweep out" in the event of a momentary deficiency of cooling which might initiate two-phase flow in the process tubes is compared with the flow available for a range of reactor power levels in Figure 4. The flow required to maintain the bulk outlet temperature at 90°C is shown by solid lines. The flow required to sweep steam out of the process tubes should two-phase flow be present is shown by dotted lines. The flow available from the flywheel decay, and after various degrees of power loss, is shown by the dash-dotted lines.

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It is seen that the flywheels are critical during the first few minutes after shut-down, and that they are not capable of preventing a modest increase in bulk outlet temperature during the first few seconds. Further, a flow of 30,000 gpm is sufficient to prevent bulk outlet temperature rise after the normal flywheel power is insufficient to provide this protection.

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APPENDIX II

BIBLIOGRAPHY ADEQUACY OF K REACTOR COOLING

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Figure 1
K-REACTOR WATER PLANT
SIMPLIFIED FLOW DIAGRAM

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Figure 2
K-REACTOR
PIPING CROSS-TIES

*ORIFICED
CHECK VALVES
A-HPCT FLOW
ORFICE

CLEAR WELL

CLEAR WELL

TRANSFORMERS

○ HIGH LIFT PUMPS (SECONDARY)

⑥ EMERGENCY GENERATOR

LOW LIFT RUMPS (PRIMARY)

Figure 3

Simplified Single-Line Schematic of The 100-K Area Power Distribution System.

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REACTOR WATER FLOW - 1000 G.P.M.

FIGURE 4
K-REACTOR FLOW REQUIREMENTS
AFTER A SCRAM, AND FLOW
BACK-UP SUPPLY VS. TIME
(INLET TEMPERATURE = 20°C)

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