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March 19, 1959

K-WATER PLANT IMPROVEMENTS

A Task Force was established within the Irradiation Processing Department to evaluate the reliability of the K Reactor Water Plants and to identify capital modifications which appear justified in securing further increments of improved K-Reactor nuclear safety. The Task Force consisted of the following:

- R.L. Dickeman - Chairman
- R.S. Bell
- H.W. Heacock
- S.S. Jones
- J.T. Long
- F.J. Mollerus
- L.P. Reinig
- G.J. Rogers
- H.G. Spencer
- R.E. Trumble

MASTER

This report summarizes information and conclusions developed in the course of Task Force deliberations.

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TABLE OF CONTENTS

	<u>Page</u>
I. INTRODUCTION - - - - -	3
II. SUMMARY - - - - -	3
A. Improvements Programmed and/or Accomplished to Secure Added Reliability - - - - -	4
B. Further Installations to Improve Reliability - - - - -	5
III. PROMINENT CONCLUSIONS - - - - -	7
IV. DISCUSSION - - - - -	8
A. Safety Criteria - Reactor Cooling System Reliability - - - - -	8
B. Description of the K-Water Plants and Their Capabilities - - - -	11
C. Operating History and Improvements - - - - -	14
D. Sequence of Events Potentially Leading to Failure of the Improved System - - - - -	21
E. Additional Capital Installations - - - - -	28

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I. INTRODUCTION

Adequate cooling of reactor fuel elements is the paramount consideration in nuclear reactor safety. This results directly from the fact that the fuel element itself, is by far the most effective and reliable fission product containment vessel devised; inadequate cooling of fuel elements inevitably leads to release of fission products. The extent of the release is a function of the details of an "inadequacy of cooling" incident but, in the case of complete loss of cooling to a reactor, could range to twenty-five percent of the total fission product inventory which has been estimated to result in one to ten billion dollars of assessable damage to life, property and crops.

While many inter-related considerations, such as reactor control and confinement can influence the quantity of fission products released to the environment in a reactor accident, these are, in fact considerations of secondary importance relative to avoiding a fuel or reactor core meltdown incident in the first place. Proper perspective in plant programs, and capital expenditures, to improve nuclear reactor safety then require that the positive action of securing, maintaining and improving reliability of adequate cooling be weighted heavily relative to passive actions such as are typified by incremental additions to control systems or provision of confinement.

This scale of values has been recognized at Hanford and prime emphasis given to sustained adequacy of cooling through the engineering, design, maintenance and operational contributions to safe reactor operations. However, it is difficult to determine that a given reactor cooling system is "reliable enough;" that is, additional incremental improvements are not justified economically when risks are balanced against costs in light of an assessed "current level of reliability." It is probable that the point at which "sufficient reliability" is judged to have been achieved will shift with increasing power levels, operating and maintenance experience, or even increased sophistication in the understanding of the nuclear safety problem and the problems in operating and maintaining complex plants.

A Task Force was established in the Irradiation Processing Department to examine the K-Reactor Water Plant to 1) review the operating and maintenance experience with the water plant as improved since startup, 2) identify major plant additions which could further improve reliability, and 3) estimate the costs of any such additions. This report summarizes information developed in the course of the Task Force deliberations.

II. SUMMARY

The several major elements of the study are summarized below with supporting detail included in the following discussion. This report also amplifies upon brief analyses previously documented.⁽¹⁾⁽²⁾

(1) HW-58614, "K Reactor Water Plant Analysis," R.L. Dickeman, 12-22-58.

(2) Confidential-Undocumented, "Reliability of K-Water Plant System," R.S. Bell, 12-16-58.

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A. IMPROVEMENTS PROGRAMMED AND/OR ACCOMPLISHED TO SECURE ADDED RELIABILITY

The K-Water Plant basically consists of the electrically driven primary cooling system with power supplied by the BPA system, electrically driven secondary or backup cooling system powered by a steam driven emergency generator pair, and a "last ditch" system consisting of hydraulic cross-ties between the two K-Water Plants.

1) Primary System

The BPA electrical power source is considered very reliable. A complete, unscheduled loss of BPA power to the HAP0 loop has never occurred and complete loss at a reactor only once - this through substation switching errors. This performance has lead to an estimate of once in ten years as a conservative frequency for complete loss of BPA power to a reactor and improvements have been and are being made to further improve the reliability of this system. The pumping and piping systems are also judged to be reliable although further work is needed, and in progress, to establish a) extent of reactor damage which would result from a pump shaft break incident at higher power levels while on five pump set operation, b) vulnerability of the system to earthquake damage, and c) flow provided by the pumps while in the instability region.

2) Secondary or Backup System

The secondary system was called upon to function during the August 21, 1955 KW incident and both emergency generators failed to assume the load. Since this time nearly \$700,000 have been spent on physical improvements to the system; the major improvements include:

- a) Adoption of a vertical split bus method of operation which secures electrical isolation between the two generators during emergency operation.
- b) Improved boiler reliability by addition of steam auxiliaries and improved air balance.
- c) Addition of automatic synchronizers to minimize switching errors.
- d) The initiation of a more complete and aggressive maintenance and testing program.

A program of testing the emergency system operation under simulated BPA power failure conditions has revealed inadequate performance once in 352 simulation tests; under these conditions the reliability at the 95 percent confidence level, of a given generator to assume its connected load is 98 percent and, one of two generators, over 99 percent.

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The reliability of the secondary system depends very critically upon the maintenance it is given. It is a complex electrical system, and, of necessity, possesses elements provided to separate it from the BPA system which are in causal relationship whereby both systems could fail simultaneously. This lack of complete electrical independence between the primary and secondary systems, combined with the complexity of the system, represent the prominent deficiencies.

3) Last-Ditch System

The "last-ditch" system, the hydraulic cross-tie system, is judged to be the least reliable of the three cooling systems - largely because of V-73 valve operation. The use of this system also risks damage to the aiding reactor and the current capacity is inadequate to carry the shutdown cooling load at elevated power levels in the summer months when the aiding reactor is shut down.

B. FURTHER INSTALLATIONS TO IMPROVE RELIABILITY

In considering capital improvements which can be made to the K-Water Plants it is desirable to have concrete objectives or criteria; this obtains because it is believed more sound to use arguments relative to reliability as a basis for possibly providing less than a considered optimum cooling system but with knowledge that "judged reliability" is serving in lieu of depth, capacity, independence or other desirable quality of an integral cooling system rather than as a basis for simply maintaining status quo. Furthermore, criteria serve as a basis for consistent judgments rendered on plants which are quite different in design and operation. The criteria used by the Task Force, "Safety Criteria - Reactor Cooling System Reliability" are given in Discussion Section A; in summary, independent primary, secondary and last-ditch cooling systems are stipulated. On the basis of this criteria the following system modifications were considered:

1) Primary System

The plant modifications planned as part of CG-775, "K-Water Plant Expansion" should include pump characteristics such that one of five pump sets can fail (shaft break, for example) without causing the remaining four sets to go into cavitation. This feature is included in the scope of the CG-775 project.

2) Secondary System

The several possible modifications of the secondary system which were identified are directed at securing 1) improved reliability of 4.16 KV power and/or greater independence of the primary and secondary systems and these are discussed as follows:

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a) Supplementary Steam Turbine Driven Pumps - \$1,000,000

A steam turbine driven pump capable of delivering 30,000 gpm would replace a service water pump in each area. The pumps would connect to the process cooling lines through check valves, the turbines which would be driven by steaming the spare boiler in one area, would normally be on standby at idling speed, and a steam cross-tie line between areas would isolate boilers on this service from those powering the emergency generators.

The cost is informally estimated at \$1,000,000 for one such pump per area or \$1,750,000 for two pumps per area. One week of reactor outage time may be required for installation and the incremental annual operating cost is estimated to be \$40,000.

The advantages are 1) an additional source of secondary system cooling is provided which is quite independent of the existing electrically driven primary and secondary systems, and 2) existing steam boilers can be used. The disadvantages are 1) the boilers must accelerate upon demand to assume the emergency load, and 2) scheduling of boiler maintenance becomes more difficult.

b) 4.16 KV Cross-tie - \$150,000

The emergency generator systems in each K Water Plant can be electrically cross-tied such that 4.16 KV power is available to power the secondary system in a stricken area following loss of BPA power and both emergency generators; the informally estimated capital cost is \$150,000.

The advantage is that an alternate source of power is available to drive a low lift pump in event of 1) loss of BPA and all 4.16 KV power in one area, 2) loss of all 4.16 KV power at one area, or 3) loss of BPA at both areas and both emergency generators (all 4.16 KV power) at one area.

The disadvantage is that care in design, fabrication, operation and maintenance is required to minimize the possibility of a fault in one plant rendering the emergency generators in both ineffective.

c) Isolation of Emergency Generators from BPA - \$2,000,000

The possibility of electrically isolating the emergency generators from BPA was also considered; in this arrangement each emergency generator would continuously power a pump set during normal operation and BPA power failure could not be communicated to the secondary system; thus, acceleration of equipment and switching of loads is not required.

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The informally estimated cost of securing this electrical isolation is \$2,000,000 for both areas, three weeks downtime per area and an annual operating cost of \$500,000 is estimated as required.

3) Last-Ditch System - \$350,000

The hydraulic cross-tie lines have operated at times with marginal reliability, currently have inadequate capacity to supply full cooling needs during the shutdown transient in the summer months with the aiding reactor shutdown and, with increases in power level, this inadequacy during the infrequent shutdowns will extend throughout the year. Up to this time the increased risk resulting from the capacity limitation has been assumed on the basis of 1) infrequency of exposure, 2) improved reliability of the secondary systems, and 3) proposed installation of the 4.16 KV cross-tie to secure further reliability in the secondary system. The last ditch system can be made more adequate and reliable through removal of flow restrictions and 1) add a high tank or 2) provide an additional hydraulic cross-tie line in parallel with the existing line; in addition, further improvements to the V-73 valve are required. Either of the above alternates can be provided for about \$350,000 with an outage time of one week per reactor required.

Either will increase the hydraulic cross-tie capacity to that capable of providing full shutdown cooling requirements following complete loss of the primary and secondary systems at one area for any power level or combination of operating conditions foreseen.

III. PROMINENT CONCLUSIONS

- A. The reliability of the K-Water Plants is judged to be good. The primary system (BPA power) is judged very reliable; the secondary system (emergency generator power) somewhat less reliable - primarily because of difficulties in maintaining the complex electrical system and the causal relationship, or lack of independence, which exists between the primary and secondary systems; and the last ditch system least reliable - primarily because of V-73 valve operating difficulties, certain capacity inadequacies and a potential risk of damage to the aiding reactor in event of hydraulic cross-tie line use or failure.
- B. Substantial improvements have been made to all three coolant systems to secure improved reliability since startup. Further improvements, short of large capital installations, have been identified as desirable and should be accomplished; for the most part this effort is in progress.
- C. Capital installations considered by the Task Force to be most attractive among the alternates developed which would further improve reliability include:
 - 1) Addition of a Steam Turbine Driven Pump in Each Area - \$1,000,000.
 - 2) 4.16 KV Electrical Cross-Tie - 150,000.
 - 3) Increased Hydraulic Cross-Tie Capacity - 350,000

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IV. DISCUSSION

A. SAFETY CRITERIA - REACTOR COOLING SYSTEM RELIABILITY

It is difficult to render judgments relative to the adequacy of a reactor cooling system without having specific criteria defining the requirements in mind. This is also necessary to assure consistency in evaluating reactor plants which are quite different in design and technology.

Criteria describing desirable characteristics of a reactor cooling system, as viewed from a nuclear safety standpoint, do not exist on a national scale. This lack of criteria for nuclear reactors generally has led to the development and adaption of criteria for use at Hanford in judging the adequacy of reactor cooling systems and in identifying cooling system improvements which should contribute to increased reliability. These criteria are intended to be applicable specifically to the Hanford production reactors although the intent, the level of requirement and certain specific requirements may be more generally applicable. As is true in any broad criteria interpretation can present a problem if not approached knowledgeably and constructively.

1) Bases

Nuclear reactor safety requires that adequate cooling be maintained at all times. The damage, in terms of loss of life and impairment of health, loss of crops and long term evacuation of lands, which would result if the fission product inventory of a Hanford reactor were released is estimated to be in the 1-10 billion dollar range. The fuel element itself is the most effective containment vessel devised and fission products will not be released in dangerous quantities as long as adequate cooling is provided.

The extreme potential liability which can accrue in event of inadequate cooling justifies unusual measures to maintain the probability of cooling system failure at a very low value. At the same time it must be recognized that the probability of failure cannot be reduced to zero; hence, regardless of the precautions taken, there will be a finite probability that a maximum potential liability accident will occur.

The ultimate reliability of a system or series of systems has not yet been expressed quantitatively in a satisfactory manner nor has an acceptable level of cooling system reliability been adequately defined. On the other hand, the mandate, "the system shall not fail" is meaningless as a basis or criterion and is equivalent to requiring that the system not be operated. Therefore, the extent of the unusual measure which will be required in a reactor plant to achieve an acceptable level of cooling system reliability will be a matter of judgment.

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Prudence requires that the failure probability be maintained at a very low level and that errors in judgment in this area be on the side of requiring "too much" rather than "too little;" normal engineering standards routinely applied to "low risk" engineering are not acceptable as bases for judgment in reactor cooling system design and operation.

The criteria presented below broadly and arbitrarily define cooling system characteristics and provisions which appear prudent to require of the cooling systems in any nuclear reactor containing a substantial fission product inventory. Failure induced by operator error, equipment failure, natural phenomenon (earthquake, lightning) are considered credible; acts of war and sabotage are not specifically considered here as appropriate bases for establishing cooling system requirements. In practice, considered evaluation of specialized situations may support responsible authorization of specific exceptions to the general requirements developed below.

2) Criteria

The normal reactor cooling system requirements shall be:

a) Three independent, reliable sources of cooling as follows:

1) Primary Cooling System

The primary cooling system will provide adequate reactor cooling during all phases of operation and shutdown. The design and operation shall be such that the secondary or last ditch backup systems are not called upon to provide the sole source of cooling during any phase of normal operation and would not be expected to be required to function as the sole supply system during the life of the plant.

2) Secondary Cooling System

The secondary cooling system shall be independent of the primary system - (common piping is acceptable) - and capable of providing adequate shutdown cooling indefinitely assuming instantaneous loss of power to the primary system. Serious consideration should be given the operation of the secondary system at partial flow capability at all times and preferably at a flow capability sufficient to satisfy the full requirements in event of primary system failure without relying upon actuation or acceleration of equipment. That flow provided by the secondary system during normal operation may contribute to the primary cooling system requirements.

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3) Last Ditch System

The last ditch cooling system must be independent of both the primary and secondary cooling systems, including piping to the reactor manifold. It must be capable of providing shutdown flow indefinitely assuming instantaneous loss of power to the primary system and concurrent failure of the secondary system to provide its rated flow.

- b) An independent system is defined as one with power sources and pumping systems such that failure or cause of failure in one cannot induce failure of the alternate systems.
- c) Adequate cooling is defined, as a minimum, as that required to maintain reactor components, including fuel elements, at temperatures below the melting point. In practice, the objective of requiring levels of cooling capable of protecting against all reactor or fuel element damage will be prudent to adopt.
- d) Failure of any single component in a system shall not be capable of rendering that system inadequate. In practice certain piping and valves may be qualified as exceptions.
- e) All engineering, construction, operation and maintenance shall meet or exceed requirements specified by applicable codes and standards.

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B. 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 stored 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 reactor. 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 Figures 3 & 4 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 5. 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 shutdown, 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.

The capabilities of various sources of coolant are summarized in the following table.

TABLE I
COOLANT SYSTEM CAPABILITIES

<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	177 163 135
2) <u>Backup</u>	Emergency Generators	6 low-lift 5 low-lift 4 low-lift	59 56 53
	(One generator only)	3 low-lift 2 low-lift 1 low-lift	50 42 30
3) <u>Last Ditch</u>	Companion Reactor (BPA 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.

**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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Pertinent observations on the adequacy of the K-Water plant complex are:

- 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 (normally about 5°C) 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 flow instability. 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 went into unstable operation. Unstable flow might 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.

Assessment of System Reliability

The reliability of the K-Water Plants cannot be expressed quantitatively due to the complexity of the system and to the multitude of quantities which cannot be accurately expressed. It is possible, however, to derive a statistical approach in expressing the reliability of some elements. The statistical approach can be misleading because factors not considered are, by implication, weighted as completely reliable; and 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 sections C and D in assessing reliability of the K-Water Plant is in a relative and qualitative sense.

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C. OPERATING HISTORY AND IMPROVEMENTS

1) Original Design Criteria

The K-Water Plant design criteria demanded that three basic requirements be met.

- a) Provide an adequate source of power for normal operation with a high degree of reliability but infrequent power interruptions permissible.
- b) Provide an emergency electrical system which would back up primary power in providing adequate power for safe reactor shutdown.
- c) Provide an adequate source of power for continuance of adequate water flow subsequent to reactor shutdown.

The design criteria were met by selection of the following systems. Primary power was supplied from BPA over the Hanford Loop which is also the source of primary power for the other Hanford reactors. The choice was based on the excellent operating record of this system. Unscheduled loss of BPA power to the Hanford Loop has never occurred. Emergency back-up generation was based on concepts which were proven by synthesis of the proposed system on a power system analyzer. The primary (BPA) and secondary (emergency generation) systems were designed so that adequate water flow would be provided to shut the reactor down safely even though two major equipment pieces failed in the primary and secondary systems. For example, any two transformers, circuit breakers, generators or busses could fail and adequate shutdown water flow could still be delivered.

A last ditch system was provided in the form of a hydraulic cross-tie between K Reactor plants. Adequacy of this system presupposed integrity of the primary electrical system in the area from which the water would originate.

2) Primary System

a) BPA Power

The continuity of flow from the primary cooling system depends 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 -

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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 HAP0 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.

Improvements made to the BPA power system during the past fifteen years are:

- 1) Increased transmission facilities throughout the network have firmed the system substantially.
- 2) Installation of fast reclosures on the 230 KV BPA system speeds up BPA restoration in 17 cycles to reduce lightning outages from two to three per year in 1945 to none in recent years.
- 3) Midway now has three buses instead of two.
- 4) There are now four lines from Grand Coulee instead of two and additional lines from the Dalles with Priest Rapids to be connected in 1959. McNary is being tied to the Bonneville lines at Goldendale and a Wanapum plant will be connected to Midway within a few years.
- 5) The general reduction of Coulee Requirements in the Portland area by connection of new Columbia River plants to this area has increased the reliability of Grand Coulee at Midway.

HAP0 230 KV loop improvements. (Total 85 miles maintained and operated by HAP0 forces).

- 1) Bonding of wood structure hardware to ground has eliminated pole fires.
- 2) The BPA-Midway-Benton line has been added as a third 230 KV leg to increase the reliability at the load center between the K plants.
- 3) Continuous reduction of operating errors with education and experience.

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The initial experience of three to four events per year causing power interruption within the HAP0 loop were reduced progressively to about 0.46 per station per year which is considered the current level. The K-Plant experience indicates six scrams during the interval February, 1955 to April, 1958 due to BPA power causes which give a statistical frequency of events causing the loop to open over this interval of 0.48 per station per year and shutdowns induced by electrical disturbances of all types of 1.12 per year per reactor. While this reliable performance serves to provide considerable comfort it is concluded that this level of reliability is not sufficient to argue for anything short of a high order of supplemental cooling reliability from secondary and backup cooling sources.

b) Pump Motors and Pumps

Failure of pump-motors and pumps have never jeopardized the adequacy of coolant flow, however several low-lift pump shafts have been replaced because of cracking. Process Limits are designed to protect the reactors in the event of a single pump shaft break.

- c) 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 further explored in studies now being initiated.

3) Secondary Cooling System

The "backup" cooling systems, the two steam driven emergency generators each coupled to selected low-lift pumps, has several times been called upon to function and has failed on one occasion (August 21, 1955). On this occasion, 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. After 43 minutes 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 5); 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 upon separation from BPA and an aggressive program of maintenance and testing undertaken,

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The response of the emergency generators to simulated BPA failures has now been tested 352 times with one failure; this indicates a statistical reliability of 98.4 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 II. The numbers shown in Table II are based on the simplifying assumption that the generators are completely independent (i.e., cause of failure of one generator could not affect the other). Consequently, the numbers in this table tend to overestimate the effects of generator redundancy and reliability.

TABLE II
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 Frequency	1/10 yrs.	2	95%	\$2 x 10 ⁹	\$ 5,000,000
	1/20 yrs.	2	95%	\$2 x 10 ⁹	\$ 2,500,000
	1/50 yrs.	2	95%	\$2 x 10 ⁹	\$ 1,000,000
	1/100 yrs.	2	95%	\$2 x 10 ⁹	\$ 500,000
Effect of Generator Reliability	1/10 yrs.	2	95%	\$2 x 10 ⁹	\$ 5,000,000
	1/10 yrs.	2	90%	\$2 x 10 ⁹	\$20,000,000
	1/10 yrs.	2	99%	\$2 x 10 ⁹	\$ 200,000
	1/10 yrs.	2	99.9%	\$2 x 10 ⁹	\$ 2,000
Effect of Disaster Cost	1/10 yrs.	2	95%	\$2 x 10 ⁹	\$ 5,000,000
	1/10 yrs.	2	95%	\$500 x 10 ⁶	\$ 1,250,000
	1/10 yrs.	2	95%	\$1 x 10 ⁹	\$ 2,500,000
	1/10 yrs.	2	95%	\$5 x 10 ⁹	\$12,500,000
Effect of Generator Redundancy	1/10 yrs.	2	95%	\$2 x 10 ⁹	\$ 5,000,000
	1/10 yrs.	3	95%	\$2 x 10 ⁹	\$ 250,000
	1/10 yrs.	4	95%	\$2 x 10 ⁹	\$ 12,500
Effect of Generator Redundancy	1/10 yrs.	2	99%	\$2 x 10 ⁹	\$ 200,000
	1/10 yrs.	3	99%	\$2 x 10 ⁹	\$ 2,000
	1/10 yrs.	4	99%	\$2 x 10 ⁹	\$ 20

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 is 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.

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A number of equipment modifications has been made to increase reliability. The 4.16 KV bus tie breakers have been operated in the open position and appropriate control revisions were completed in order to provide maximum electrical isolation of emergency generators. Steam driven auxiliaries have been added to the boilers to supplement electric drives. Ventilation improvements have been made to maintain positive pressure to the boiler forced draft fans. Automatic synchronizers have been added to the power system controls to preclude human error in connecting and disconnecting critical loads. Modifications completed during the last three years to improve reliability of the K-Water Plant have cost \$676,147. The major improvements are tabulated below in Table III.

4) 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 fifteen 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.

Flow capacity of the combined low and high pressure cross-tie lines is 18,000 to 19,000 gpm with low-lift pumps in service and approximately 30,000 gpm with both high and low lifts in service.

August 21, 1955 Incidents - The cross-tie system functioned as designed and delivered flow to KW Reactor twice on August 21, 1955. On both occasions power for low lift pumps had been lost; once while the reactor was operating and a second time (while testing) after the reactor had been shut down approximately 17 hours. In no case, including these incidents, has the last ditch system been called upon to provide the only source of coolant supply.

Reliability of the system to deliver rated flows has been high for KW Reactor. For KE, due to V-73 valve problems, it has varied from moderate reliability at start-up to almost total unreliability in June and July, 1956 to the present high reliability it has demonstrated since August, 1957.

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TABLE III

IMPROVEMENTS TO THE K EMERGENCY WATER SUPPLY SYSTEM

Title	Date Completed		Cost	
	KE	KW	KE	KW
Ventilation Improvement, Buildings 165 and 190-KE and KW	4-3-58	4-3-58	\$143,621	(both areas)
Steam Driven Auxiliaries	3-28-58	3-28-58	\$434,400	(both areas)
Automatic Synchronizers, 165-KE and KW	2-5-59	11-30-58	\$ 71,419	(both areas)
Design Change No. 120 - Modification of 4160 V Switch Gear Control to Permit Testing of Emergency Generator Facilities - 165-KE and KW Buildings	7-26-56	8-22-56	\$ 758	\$ 758
Design Change No. 194 - Safety Circuit Trip Modifications Process Pump Cross-tie Valve Safety Circuit Trip Additions	Scheduled for 4-15-59	11-30-58	\$ -	\$ 4,060
Design Change No. 196 - Cross-tie Valve Control Modification	11-30-58	6-28-58	\$ 313	\$ 313
Design Change No. 155 - Safety Circuit Revisions	Not Inst.	11-30-58	-	\$ 476
Design Change No. 184 - Secondary Pump Motor, Rotor Protection Relay - 100-KE and KW	Not Inst.	Not Inst.	-	-
Replace Time Delay Relays in 480 V Station Service With Precision Switch Gear Type Time Delay Relay	5-3-56	5-8-56	\$ 101	\$ 101
Disconnect over-frequency relays from 86K relay - 165-KE and KW	8-55	8-55	None	None
Provide throw-over switches to DC power supplies for power system control - 165-KE and KW	11-55	11-55	\$ 500	\$ 500
Purchase oscillograph for power system testing	11-14-57	11-14-57	\$ 18,827	(Both areas)
Grand Total			<u>\$676,147</u>	

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Quantitative Adequacy of the system is limited to certain operating power levels as presented below:

CROSS-TIE POWER LEVEL LIMIT

<u>Type of Charge</u>	<u>Number of Operating Pumps</u>	<u>Initial Reactor Flow (1000 gpm)</u>	<u>Limiting Power Level, MW</u>	
			<u>for 0°C Inlet</u>	<u>for 20°C Inlet</u>
I & E	5	163	3800	2900
KII or KIII	6	178	3900	3000

Operating levels exceeded a cross-tie limit during July, August and September of 1957 and 1958. While operating under these conditions, direct dependence was placed upon one of the two required emergency generators remaining in service to power low lifts for emergency flow. Reliability of each generator appeared to be very high as indicated by successful results of numerous tests which removed BPA power from low lift pumps and required generators to assume the pumping load.

Modifications indicated and action taken - Last Ditch System.

Problems arising to adversely affect reliability of the last ditch system have been eliminated or minimized as listed below:

(Presentation is in probable order of greatest increase in reactor safety)

Problem

Corrective Action

1. KE V-73 valve sticking due to improper valve seating adjustment and inadequate lubrication. Established optimum seating adjustment on KE V-73 as recommended by manufacturer. V-73 tested and lubricated more frequently at both reactors.
2. V-73 air exhaust valve sticking and cross-routing "opening" air through exhaust lines to "closing" side of air cylinder. Dual-action exhaust valve CTB was replaced with two single action valves and exhaust lines physically separated at both reactors.
3. Air in high pressure cross-tie system. Introduced into system when flushing and when making line modifications. Flush valve motors interlocked with 165-105 emergency supply line isolation valves (V-70). Vent line installed on high pressure cross-tie line at both reactors.

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Problem

Corrective Action

4. V-73 closed too fast. Caused severe water hammer stress on cross-tie (Gave Ball 3X drop at each reactor).	Slowed closing time of V-73 with adjustment of air supply needle valves. Later installed orifices to control valve closing time.
5. Rust from cross-tie system plugged K Reactor supply cross-header screens.	Increased cross-tie flushing frequency.

D. SEQUENCE OF EVENTS POTENTIALLY LEADING TO FAILURE OF THE IMPROVED SYSTEM

The performance of the K-Water Plant thus far leads to the following general observations:

- 1) The primary source of power, BPA, is reliable and the estimates would place the frequency of complete unscheduled loss of BPA power to a reactor at once in not less than ten years.
- 2) Although the steam driven emergency generator secondary system failed to deliver any flow on August 21, 1955, many improvements in the system and its maintenance have since been made. The improved individual generator system has failed to pick up the load but once in 352 simulated BPA power failure tests to yield an indicated generator reliability of over 98 percent at the 95 percent confidence level. It is further observed that 1) reliability of this system is strongly coupled to level and competence of maintenance it receives, and 2) simulated failure tests cannot reflect entirely the causal inter-relationships which exist between the primary and secondary systems wherein the same sequence of events affecting one system may impair the operability of the other.

In view of the above a quick review was made of the integral systems in an attempt to deliberately identify various events or sequence of events which could render one or more elements of the cooling system inoperable. An attempt was also made to estimate the frequency of failure and the consequences. Time available has not permitted the detailed evaluation of each failure event, estimated failure rate, mode of failure or credibility of a given failure mode. The elements identified are given below in Table IV and are useful in establishing relative magnitudes and bases for further detailed study. The following remarks will assist in putting Table IV in proper perspective:

The 230 KV System

In order to appreciate the meaning of BPA failure, events of failure which would cause total loss of 230 KV power at the K Plants, must be

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traced from the Midway Sub-station to K Plant. Any one of the following simultaneous occurrences would cause total loss of 230 KV power at K Plant: (1) lose 7 power sources to Midway Sub-station, (2) lose 3 busses at Midway Sub-station, (3) lose 3 (2 when maintenance is being performed on one line) transmission lines to K Plant from Midway Sub-station, (4) 2 faults - one on either side of a K Plant, and (5) 2 operator errors which cause the 230 KV sources to be switched off.

Credible Failure of Power Supplies, Pumps and Pump Discharge Lines

There are several sources of causal relationships between the primary and secondary systems. The two systems are common in that the 13.8 KV and 4.16 KV switch gear occupy the same room. Inter-connecting cables occupy the same cable vault and all pump motor cables are common in a confined region within the low pressure cross-tie tunnel immediately above pump discharge line numbers 1, 2 and 3. It is believed that no disaster short of a major earthquake could simultaneously reduce all of the above equipment to ineffectiveness. Occurrence of fire appears to be incredible within this structure since it is built of steel and concrete; the equipment is principally fabricated of metal and wire whose insulation is not particularly combustible.

The primary and secondary systems become more intimately related when viewed from the aspect of protective relaying. D. C. control power which supplies protective relays in the 230 KV, 13.8 KV and 4.16 KV systems emanates from a single metal clad control panel which in turn is supplied by two independent battery systems each housed in a separate room. Failure of this control power would leave all of the electrical systems incapable of fault clearance. Protective relays and circuit breakers could not operate since they require "power on" to function and therefore are not fail safe in this sense.

Power system studies show that generator stability can be maintained subsequent to faults if clearing time is maintained at less than 20 cycles for a sustained under voltage of 75%. Each generator is presently operated independently of the other at the 4.16 KV level. The generators are fitted with separate protective relays each of which is capable of separating its respective generator from the primary system in the event of serious distress. Failure of one separation relay set would not effect the ability of the other set to function properly. It is absolutely necessary that one of the two under voltage relay sets function properly during sustained fault conditions at 75% voltage for 20 cycles on the primary system in order to retain adequate secondary system power necessary to maintain reactor shutdown flow.

The following sequence of events would lead to simultaneous failure of the primary and secondary systems.

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- a) Fault on 230 KV system, time equals zero cycles (60 cycles per second base)
- b) First line 230 KV line fault protection fails, time equals zero plus 6 cycles.
- c) Back-up 230 KV fault protection fails, zero plus 12 cycles.
- d) Both KW under voltage relays fail, zero plus 15 cycles.
- e) Any time after zero plus 20 cycles the secondary system will probably be lost as a result of generator instability even though separation might be achieved in 25 or 30 cycles. After 30 cycles of sustained under voltage (75%) the high lift pump motors would be tripped by under voltage relays.

Process Line Rupture (Headers, Risers, Cross-Tie Lines)

In the analysis of the effects of line rupture a 50% of pipe free cross-section was assumed as the area of the rupture in the form of a nearly round opening. This is a very severe assumption and study is under way to obtain a more realistic evaluation of the rupture equivalent diameter and cross-section. Using this arbitrary 50% of free cross-section the effect of a rupture on process pump capacity has been estimated from pump characteristic curves. (Figure 6) In the event of a riser rupture, for example, essentially all of the flow to that riser would be lost. The remaining flow to the reactor would depend upon the location of the rupture, the status of the FB-5 valve, and the effect upon the pump flow rate of the reduced back pressure. The worst consequence of such a rupture would be a reactor melt-down. However a severe earthquake would be required to produce a riser rupture of this consequence.

The frequency estimations in Table IV differ markedly from those presented in Table II chiefly because of a marked difference in the concept of a failure sequence. The event sequences in Table II are assumed to be simultaneous unrelated events whose frequency diminishes as f^x . Where f is the failure frequency (number per year) for one generator and x is the number of generators. In Table IV a more pessimistic estimate of the failure frequency results from consideration of the causal relationship between the various parts of the K pumping and power system.

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TABLE IV
WATER PLANT FAILURES, CREDIBILITY AND CONSEQUENCES

<u>Failure Event</u>	<u>Estimated Failure Rate</u>	<u>Credibility</u>	<u>Damage To</u>	
		<u>Modes of Failure</u>	<u>Reactor</u>	<u>Effluent Piping</u>
1. 230 KV on Hanford Loop	1/(10 yrs.)	<ol style="list-style-type: none"> 1. Lose 7 transmission lines to Midway sub-station 2. Lose Midway sub-station by fire, earthquake, flood, wind 3. Lose 3 or possibly 2 (during maint.) transmission lines to Hanford from Midway 4. Two simultaneous faults on 230 KV loop, one on either side of a K Plant 5. Sub-station operator at K Plant erroneously trips off both sources of 230 KV 6. Sub-station operator at K Plant trips off remaining source of power after other is lost 	None	None
2. 2 Emergency Generators; 4.16 KV System Energized from Hanford Loop	1/(50 yrs.)	<ol style="list-style-type: none"> 1. Control operator erroneously trips off both generators 2. Pilot exciters fail on both generators 3. Main exciters fail simultaneously on both generators 4. Voltage regulators fail simultaneously on both generators 5. Turbines fail simultaneously on both generators 6. Steam generators fail simultaneously on both generators 7. Protective relays fail simultaneously on both generators 8. Electrical faults occur simultaneously in both generators or generator power circuits. 	None	None

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TABLE IV (CONT.)

<u>Failure Event</u>	<u>Credibility</u> <u>Failure Rate</u>	<u>Modes of Failure</u>	<u>Damage To</u>	
			<u>Reactor</u>	<u>Effluent Piping</u>
3. 4.16 KV System; 13.8 KV System Energized by Hanford Loop	1/10 yrs.	<ol style="list-style-type: none"> 1. Four simultaneous or cascading bus faults 2. Simultaneous or cascading failure of 2 transformers plus 2 generators 3. Simultaneous false tripping of 4 sets of protective relays 4. Operator erroneously trips off 2 transformers and 2 generators 5. Operator erroneously trips off power to B and E busses after it is lost to C-F through generator and relay failure 6. Separation of BPA from 4.16 KV system followed by immediate generator loss 	None	Minor at 93° Bulk Moderate at 96° Bulk
4. 230 KV plus one generator	1/15 yrs.	Modes of generators and 230 KV listed above	None	None
5. 230 KV and 2 generators at one area; other area operating	1/40 yrs.	Generators and 230 KV modes of failure listed above	None	Possible minor effluent piping damage to aiding reactor at 93° bulk Moderate to aiding reactor at 96° bulk, no damage at stricken reactor
6. BPA and 2 generators; other area down	1/50 yrs.	Generators and 230 KV modes of failure listed above	None	Possible severe effluent damage to stricken reactor at either 92 or 96° bulk (test data incomplete)

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HM-59696

TABLE IV (CONT.)

<u>Failure Event</u>	<u>Failure Rate</u>	<u>Credibility</u>	<u>Modes of Failure</u>	<u>Damage to</u>	
				<u>Reactor</u>	<u>Effluent Piping</u>
7. Loss of Pump without benefit of flywheel effect			1. Failure of pump shaft between pump and flywheel		
a. Primary or secondary pump; (6 pump operation)	1/8 yrs.		2. Cavitation failure of one impellor blade causing loss of all blades weakened by cavitation	7a None	Minor
b. Primary or secondary pump; (5 pump operation)	1/8 yrs.		3. Failure of pump casing.	7b Severe	Severe
8. Loss of Reactor Front Riser	Negligible		1. Almost incredible could only be caused by very severe earthquake.	Would range from moderate to complete depending upon the location of the rupture and the position of the FB-5 valve.	
9. Loss of Process line between the 165 Bldg., valve-pit and the reactor risers	Very low 1/50 yrs.		1. Severe earthquake		
a. 6-pump operation			2. Fatigue weld failure from excessive vibrations.	9a Minor to severe depending upon FB-5 valves	Severe
b. 5-pump Operation				9b Complete	Severe

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TABLE IV (CONT.)

<u>Failure Event</u>	<u>Failure Rate</u>	<u>Credibility</u>	<u>Modes of Failure</u>	<u>Damage To</u>	
				<u>Reactor</u>	<u>Effluent Piping</u>
10. Loss of Pump lines (clear well to pump discharge check valve)	Very low ~1/50 yrs.		<ol style="list-style-type: none"> 1. Severe earthquake 2. Water hammer from sudden closure of stuck check valve. 3. Weld failure by fatigue from severe vibration. 	Same as the loss of a pump	
11. Loss of high pressure cross-tie			<ol style="list-style-type: none"> 1. Severe earthquake 2. Water hammer from sudden closure or opening of a V-73 valve. 		
a. Line failure between V-72 valves	~1/40 yrs.		Same as above	Minor	Severe
b. Line failure between reactor and V-72 valve			Same as above	Moderate	Severe
c. Failure of V-73 valve			<ol style="list-style-type: none"> 1. Loss of all power at one reactor, V-73 valve does not open 2. Loss of power to down reactor V-73 does not open. 	Complete None-Moderate	Complete Minor

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E. ADDITIONAL CAPITAL INSTALLATIONS

Review of the operating history, system improvements and failure potential of the K-Water Plant indicates that the primary, secondary and backup systems are adequate in depth and that these systems, properly maintained, can be classed as reliable.

It is recognized that reliability is a relative term and that the potential catastrophic consequences of coolant system failure place extraordinary demands upon the reliability of the K-Water Plant coolant supply systems. In view of the causal relationships between the primary and secondary systems which might cause some degree of sequential failure, in view of the limited capacity of the hydraulic cross-tie lines, and considering the difficulties inherent in adequately maintaining complex systems, additional installations to improve reliability of coolant supply are considered.

Six alternate installations were considered. For each installation, the following section contains a brief description, a preliminary estimate of capital and annual operating costs, and a summary of advantages and disadvantages.

Three systems are considered worthy of further study, and these are discussed in the second part of this section. Primary attention is given to the increment of improved cooling system reliability provided by each, particularly with respect to the potential failures discussed in Section D above and the degree of conformance to the Proposed Criteria for Reactor Cooling System Reliability.

1) Alternate Installations

a) 4.16 KV Electrical Cross-tie⁽³⁾

This scheme provides an auxiliary 4.16 KV bus in each area which receives power from either of the emergency generators in the other area and which can supply power to either No. 3 or No. 4 low lift pump, but not both at once. The estimated capital cost is about \$150,000. The electrical system is shown schematically in Figure 7.

The advantages are to provide an alternate power supply to a low lift pump in the event of either 1) loss of both BPA and all 4.16 KV power at one area, 2) all 4.16 KV power at one area, or 3) loss of BPA at both areas and loss of all 4.16 KV power at one area.

The disadvantages are that the emergency power system in each area may be further exposed to fault potential, and additional maintenance will be required to service the new breakers and relays; the significance of these will depend upon detailed design.

b) Isolation of Emergency Generator - Low Lift Pump Sets

One emergency generator and one low lift pump would be connected on each of two busses in each area. Each generator-pump set would be physically as well as electrically isolated, and would be in continuous operation. Alternate pumps could be transferred to each generator during outages to permit maintenance.

The estimated capital cost is about \$2,000,000 for both areas, and about three weeks downtime per area is required for installation. Annual operating costs are estimated at about \$500,000 per year for both reactors.

The advantages are 1) passive transition on loss of BPA, 2) failure of one system cannot affect the other, and 3) no possibility of generator overload.

The disadvantages are 1) failure of any generator pump set would probably scram the reactor, 2) closer surveillance of voltage and frequency drift would be required, 3) peak load trimming and power factor correction capabilities would be lost, 4) annual operating costs are increased considerably, and 5) loss of 1 pump yields loss of 1/2 of backup.

c) Supplementary Steam Turbine Driven Pumps

This scheme would replace one or two of the service water pumps in each area with a 30,000 gpm steam turbine driven pump. These pumps would connect to the reactor process lines through check valves. The turbines would be driven by steaming the spare boiler in one area. A steam cross-tie line between areas would be provided to isolate the boilers on this service from those powering emergency generators. The turbine driven pump would be on standby at idling speed.

The capital cost for one such pump per area is estimated at \$1,000,000. Two such pumps per area would cost \$1,750,000. About one week per reactor downtime for installation would be required. The annual operating cost is estimated to be \$40,000 for both areas.

The advantages are 1) an additional non-electric source of cooling is provided, and 2) an existing boiler is used.

The disadvantages are 1) the boiler must accelerate to pick up the emergency load, and 2) scheduling of boiler maintenance becomes more difficult.

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d) Hydraulic Cross-Tie Line Modifications

This proposal has two alternates which appear to cost about the same; 1) modify the existing high pressure cross-tie line to remove flow restrictions and add a high tank to provide greater transient flows, or 2) provide an additional cross-tie line in parallel with the existing line.

The capital cost of either alternate is estimated at about \$350,000 and an installation outage of about one week per reactor would be required.

The advantage is to make the cross-tie system adequate for any total pumping power loss at one area.

The disadvantage is that a hazard may exist for the aiding reactor in the event of certain process line breaks at the stricken reactor or in the cross-tie line itself.

e) Supplementary River Pump Power Supply

This scheme would provide diesel driven generator sets to supply emergency power to two of the pumps at each river pumphouse. The raw water could be supplied either to the filter plant or to the reactor via existing piping in the event of total power loss at both areas.

The capital cost is estimated to be about \$750,000 for both areas, and no installation outage would be required. Annual operating costs are estimated at about \$10,000 per year.

The advantage of this system is to provide a long term raw water supply which is independent of existing electric power supplies.

Its disadvantage is that it would not be available for at least 5 minutes after power loss because the ball valves on unused pump discharge lines must be closed.

f) Portable River Water Pumps

This scheme would provide truck mounted diesel driven pumps and invasion piping which could be set up in any area to replenish raw water reservoirs if all river pumps were lost for any reason.

Such equipment would cost an estimated \$150,000.

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2) Installations Suggested for Further Consideration

a) Hydraulic Cross-Tie Line Modifications

The hydraulic cross-tie line is the only installation which satisfies the proposed criteria as a last ditch cooling system, that is, one independent of the primary and secondary system as to pumping energy, water supply, and piping to the reactor. Accordingly, the modifications to make this system adequate for foreseeable operating conditions seem to be well justified if it is agreed that an adequate last ditch system is required regardless of the reliability of the primary and secondary cooling systems.

There is some uncertainty regarding the flow capability of the existing cross-tie lines, and the adequacy of the existing lines at future operating levels cannot be confidently stated. It is therefore suggested that tests be performed to ascertain the exact extent of the modifications required to this system to support future conditions. It is believed prudent to budget for the modifications described previously, without regard to any of the following items.

b) Supplementary Steam Driven Pumps

This proposal is basically an additional secondary cooling system. It serves as an additional step between the existing electrical powered secondary system and the last ditch system. It meets the proposed criteria for secondary systems, although it is not furnishing flow to the primary system continuously. It is disqualified as a last ditch system because it would not use separate piping to the reactor, and its basic energy source (boilers) is common to the electric powered secondary system though they would be isolated by valving. Its prime justification would seem to rest on lack of independence between the primary and secondary systems. It is recommended for further study and possible budgetary action on the basis that it is the only proposal considered which is available at reasonable cost in the event that the existing electric powered secondary system is judged to be of inadequate reliability.

c) 4.16 KV Cross-Tie Between Reactor Plants

The 4.16 KV cross-tie represents a method of greatly increasing the reliability of the secondary cooling system; the emergency generators from the companion plant can then power selected secondary pumps in the stricken plant. Caution in the design must be given to ensuring that faults in the stricken plant cannot be communicated through the electrical tie to reduce the reliability of the emergency system in the aiding plant. This system should be studied further and a possible approach is shown in Figure 7.

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FIGURE 1
K-Reactor Water Plant
(Simplified Flow Diagram)

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Figure 2
K-REACTOR
PIPING CROSS-TIES

XORICED
CHECK VALVES
Δ - HPCT FLOW
ORFICE

Handwritten signature and date: 12-23-59

FIGURE 3

230 KV SYSTEM
(SIMPLIFIED DIAGRAM)

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LOW LIFT PUMPS (PRIMARY)

Figure 4

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

FIGURE 5

**K-Reactor Flow Requirements After a Scram, and Flow Back-up
Supply vs. Time (Inlet Temperature = 20°C)**

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Figure 6: K REACTOR PUMP AND FUEL FLOW CHARACTERISTICS

FIGURE 7

4.16 KV ELECTRICAL CROSS TIE BETWEEN 165 KE & KW

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