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RELIABILITY AND SAFETY OF THE K REACTOR COOLING SYSTEM:

PART II

ENGINEERING ANALYSIS OF HYDRAULIC AND MECHANICAL ASPECTS

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by

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April 4, 1960

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INTRODUCTION

- 1.0 Subsequent to the recent formulation and adoption of safety criteria for reactor cooling systems, there appeared the need for an independent evaluation of the safety and reliability of the K-Reactor cooling system in terms of these criteria. The primary, secondary, and last-ditch cooling systems of this reactor involve a strong inter-dependence between electrical and hydraulic components of the water plant. Because of the complexity of inter-relationships between these components, the analysis was divided into two parallel studies which were accomplished during the summer of 1959. F. D. Robbins has presented his analysis of the electrical power and control system in HW-61887(5). This report deals with an engineering analysis of the hydraulic and mechanical aspects of the reliability and safety of the K-Reactor Cooling System. The system, as described in this report, is that which existed during the summer of 1959, prior to modification under Project CG-775 (now Project CG-883).

SUMMARY

- 2.0 A thorough study was made of the water plant through examination of "as-built" drawings, physical inspection of the facilities, study of actual operating data, and conference with engineers and operating personnel closely associated with the system. In addition, numerous documents pertaining to the system were reviewed, and data from many of these are summarized in this report.

In his approach towards an evaluation of the system, the writer formulated a comprehensive list of investigations required for the study (Appendix D), and then selected from this list those items considered most appropriate for the limited time available. Subsequently, an attempt was made to delineate the most important of safety and reliability problems existing in the areas studied. These areas are listed briefly as follows:

- a. Primary cooling system (pumps, piping, and valves).
- b. Secondary cooling system (piping system and steam plant).
- c. Last-ditch system (high-pressure crosstie piping and valves).
- d. Operation and control of water plant.

CONCLUSIONS AND RECOMMENDATIONS

- 3.0 Conclusions drawn from this analysis of the reliability and safety of the K-Reactor Water Plant are presented in essentially two groups in the following paragraphs. The first of these deals with potential coolant flow inadequacies caused by mechanical or operational failure of major components of the system, while the second concerns the extent to which the reactor cooling system safety criteria are satisfied. The actual significance of the potential failures and inadequacies, with respect to reactor safety, is dependent upon the reactor damage they might cause. This aspect

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was not within the scope of this analysis; consequently, no interpretation is given in connection with the relative degree of safety hazard associated with each of the postulated failures or flow deficiencies.

The following situations could result from mechanical or operational failures of major cooling system components:

- a. Possibility of sudden flow reduction or stoppage caused by failure of one pump with loss of flywheel action. Such an event could be caused by failure of pump shaft between impeller and flywheel or other factors, and possibly could result in failure of the check valve in the process header leading from the affected pump set, or rupture of the process header itself. The magnitude of flow reduction was estimated to range between eight percent for 6-pump operation prior to the failure to almost complete loss of water.
- b. Possibility of sudden flow reduction caused in an operating reactor by the inadvertent opening of the V-73 valve at the neighboring reactor at a time when it was down. The magnitude of this flow reduction was estimated as ranging between seven to thirteen percent, according to the mode of operation of the affected reactor prior to the incident.
- c. Possibility of high pressure crosstie rupture due to water-hammer caused by too-rapid opening or closing of the V-73 valve. Large reductions in reactor flow could be experienced; the magnitude depending upon the nature and location of the rupture. Almost complete loss of water to one side of the reactor would result if the rupture occurred at or near the connection to the A-B risers.⁽¹⁸⁾
- d. Flow deficiency during a scram caused by BPA interruption because of inadequate capacity of the high pressure crosstie. It was found that there appeared to be at least two credible cases in which the high pressure crosstie was inadequate in providing transient flow requirements for extended periods, after the flow capabilities of the pump set flywheels had been expended. The maximum magnitudes of the deficiencies ranged between 9 and 44 percent of the required flow at a 3400 MW power level.
- e. Possibility of inadequacy of the high pressure crosstie because of failure of the V-73 valve to operate. The maximum magnitude of flow deficiency was estimated at approximately 44 percent of the required flow at a 3400 MW power level.

On the basis of all considerations involved, the following conclusions are drawn relative to specific areas in which the criteria for safety of reactor cooling systems⁽¹⁴⁾ are not satisfied. (See section 4.4.1 for system classification.)

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Primary Cooling System - This system is deficient because of its susceptibility to failure of a single component. Varying degrees of flow deficiency could result from failure of a pump with loss of flywheel action, inadvertent opening of the V-73 valve at the neighboring reactor, or water-hammer rupture of the high pressure crosstie.

Secondary Cooling System - This system is not independent of the primary system and, in addition, loss of flywheel action upon failure of a pump shaft could cause effective temporary failure of the system.

Last-Ditch System - This system is not independent of the other two because the last-ditch system as referred to one reactor is dependent upon the primary and secondary systems of the other reactor. The primary systems of the two reactors, the failure of one of which could require operation of the last-ditch system, are electrically dependent upon one another (due to the 13.8 KV intertie). In addition, failure of a single hydraulic component as described previously could cause total failure of the system, if the contribution of the low pressure crosstie is ignored.

On the basis of the limited analysis conducted in this study, all of the foregoing possibilities appear to be credible. It is recommended, however, that further studies be made not only to determine the degree of credibility of each of the events (particularly the consequences of pump failure with loss of flywheel action), but also to estimate the extent of reactor damage and public safety-hazard aspect associated with each of the events. Although considerable work has been done in this respect, review of other writings of the subject has indicated that much more study is required; particularly in connection with the consequences of pump failure.

It appears that some remedial action is required in order to provide for operation within the doctrine as set forth in the safety criteria. General recommendations to modify the three systems of the K Water Plant or to alter their modes of operation are given in the following paragraphs; specific recommendations in this respect are given in Section 4.7.

- a. Primary Cooling System - The principal action required is in reducing the relative magnitudes of flow deficiencies which might occur as the result of the previously-discussed events. This may be accomplished by provision of an emergency reserve of high-pressure cooling water at the reactor risers and by altering the manner of operation of the process pump sets (coupled with crossheader throttling). Flow deficiencies resulting from failure of a pump with loss of flywheel action can best be minimized by increasing the number of operating pump sets; i.e., adding pumps to the existing system.

- b. Secondary Cooling System - Electrical dependence between primary and secondary cooling systems appears to rule out the possibility of making any modifications to the latter to provide satisfaction of the safety criteria. A separate pump (or group of pumps), driven by a completely independent power source, such as proposed in CGI-844(32,33) would remove the dependency caused by the use of common pumping equipment to serve the two systems.
- c. Last-Ditch System - Deficiencies of this system are in its inadequacy to provide sufficient coolant flow in several cases initiated by BPA separation, its dependence upon the other two systems, and the lack of reliability of the V-73 valve. The minimum flow capability of this system should be increased to a level substantially above the minimum requirement by means such as outlined in HW-61045(33) (re: CGI-844), and should be kept in step with all future reactor power-level increases. If this remedial action, and those recommended in the foregoing paragraphs for primary and secondary systems, is taken, then the last-ditch system will be independent of these systems, providing that operation of the V-73 valve can be made reliable. As discussed in detail in section 4.7, the operating mechanism of this valve should be modified so that the opening and closing time can be regulated with greater precision.

General recommendations are given in Section 4.8 with respect to the following:

1. Problems requiring further analysis.
2. Equipment tests to improve reliability.
3. Modes of operation of the water plant to increase reliability.
4. General control and operation.

DISCUSSION

4.1 DESCRIPTION OF THE WATER PLANT

The 100-K water plant as described in this report includes facilities which supply all important water requirements associated with the reactor. These facilities are, essentially, the process water system and the service water system. An additional system, which supplies the raw water requirements of the plant, will be discussed only in limited scope, since its operation and control have little bearing on the reliability of the water plant from the standpoint of reactor hazards.

Figure 1 shows the principal water lines leading to the K-West reactor and this will be used as a reference in the following description of the plant. Insofar as this study is concerned, the K-East plant is identical. The flow diagram was prepared from as-built drawings, a list of which is given in Appendix C, and was verified by visual inspection of the installed piping.

Raw water is supplied from the 181 Building river pumphouse containing six vertical pumping units, which discharge into two 60-inch raw water headers. These headers run through the 165 valve pit, to the 183 head house and thence to the filter plant. One butterfly valve in each header (RW-1 and RW-2 in Figure 1) may be closed so that emergency raw water may be supplied through strainers to the process water system, the service water system, and the condensers in the steam plant.

The function of the process water system is to supply normal cooling water to the associated reactor and, in some cases, emergency flow to the neighboring reactor. Water requirements of the filter plant, process pump cooling, boiler feed makeup, and other plant demands are supplied by the service water system. These requirements will be summarized in detail in a later section.

Filtered water to serve all normal plant requirements is supplied from two clearwells, having a combined capacity of 18.25 million gallons*, which are connected to the two 190 building pumpwells shown in Figure 1. These pumpwells are interconnected by a cross-tie tunnel having normally-open sluice gates at both ends. Three openings in each pumpwell admit water through normally-open sluice gates from the associated clearwell. The clearwell floors are four feet above the pumpwell floors.

Three process pump sets, two service water pump sets, and one backwash pump draw water from each pumpwell. The backwash and service water units are associated only with filter plant operation and operate intermittently; no description of their system will be included here.

Process Pump and Piping System

The six process pump sets are operated in parallel and discharge into the 24-inch process headers shown in Figure 1. In each process pump set, a low-lift mixed-flow unit draws from the pumpwell and is connected in series with a high-lift centrifugal pump. Characteristic curves for each of these pumps are given in Figures 2-5, both as originally installed and as they will be when modified under Project CG-775 (now Project CG-883). At present, one pump set (Unit No. 1 in KE) has been modified under the K expansion program. Rated operating characteristics are shown in Table 1 and combined pump set and reactor system characteristics are given in Figure 6 for the original pump sets.

* Including pumpwell storage above zero clearwell depth.⁽¹⁾

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TABLE 1
RATED PUMP OPERATING CHARACTERISTICS

Pump	Pre CG-775		Post CG-775 (1)	
	Low Lift	High Lift	Low Lift	High Lift
TDH-ft.	122	912	135	950
Capacity-GPM	28,000	28,000	35,000	35,000
Pump RPM	588	1,696*	588	1,696*
Pump and Fly-wheel WR^2	77,000	156,000		

(1) HW-60659 (Ref. 2)

* Maximum

Each primary pump is directly connected to its motor, while the secondary pump is driven through a geared speed increaser. Pump set capacity control is effected by use of a wound-rotor induction motor (high lift drive) connected through a liquid rheostat. Control of capacity of the low-lift pump (when operated alone) is accomplished by throttling the pump discharge (PD) valve in the 24-inch process header. Both pumps in the set are provided with flywheels to provide gradual flow decay upon loss of power.

The primary function of a low lift pump is to provide enough pressure at the suction of the associated high lift unit so that damaging cavitation will not occur in the latter under design conditions of operation. This arrangement makes it possible to operate the low lift pumps separately, at lower discharge pressure and discharge, to provide shutdown flow to the reactor. Under these conditions, the low-lift pumps discharge through the bypass lines shown in Figure 1, which lines are provided with check valves.

A recirculation pipe from the discharge piping of the pump set is connected through a three-way valve to the service water loop and to the condensate return line leading from the steam plant. An eight-inch test discharge line is located in each process header just upstream from the main discharge valve.

From the 190 pumphouse, the 24-inch process headers run to the 165 valve pit where they join into two 36-inch pipes leading to the risers of the reactor (in Figure 1 the A and B, and C and D headers were reversed to simplify the schematic). The two outboard sets of headers have only pump discharge (PD) valves, which are lubricated plug-type and motor operated, and check valves. The discharge pipes of the two inboard pumps (No. 3 and 4) are valved to the 36-inch headers through four flow balancing valves (FB-1-4) and, in addition, are cross-connected by a 24-inch line containing

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a sectionalizing valve (FB-5). The flow balancing valves (FB-1-5) are provided to balance the flow to all risers according to the mode of operation of the pump sets. It will be noted that the A-B risers are tied together, with check valves upstream from the tie line, while the C-D risers are not.

Service Water Pump and Piping System

The service water pumps are connected through check and discharge valves into a 24-inch service water loop which supplies 183, 165, 190 and 105 building water requirements. Valves are arranged in the system so that the various portions of the loop may be sectionalized. A characteristic curve for a service water pump is given in Figure 7; each unit is rated at 4000 gpm at a 255 ft. TDH.

Emergency Interconnections and Crossties

A number of interconnections between the various systems are made in order to provide emergency water in case of inadequacy or failure of a given system. Most of these connections are made in the 165 valve pit; the single exception being a crosstie to provide emergency process water to the opposite reactor. These systems will be discussed briefly in the following paragraphs; operating details will be given in a later section.

Raw Water Inter-ties:

By closing the main raw water valves RW1 and RW2 in the 60-inch raw water lines, provision is made to supply emergency raw water through strainers and into the process water system (through RCE 1 and RCE 2) when process water pressure drops below a maximum raw water pressure as established by relief valves. Flow is also made available to the condensers of the power plant through RWC or to a low pressure crosstie connecting the KE and KW service water loops. This crosstie normally is supplied by the service water loop and connects to the loop of the opposite plant at a similar location.

Service Water Inter-Ties:

Emergency water from the service water loop can be supplied to the process system through the normally-open valves SE-1 and SE-2 in any case when the check valves open due to a drop in process water pressure to a level below that in the service water loop. An additional inter-tie, not shown in the flow diagram, provides a source of emergency water for the boiler feed-water supply. A direct connection to the low pressure crosstie through normally-open SC-1 provides for supply of water to the service water system from that of the neighboring reactor.

Process Water Inter-Ties:

Process water is available under emergency conditions to the service water loop of the associated reactor and to the neighboring reactor. The process to service water inter-tie consists of two 6-inch pipes, bypassing the valves

SE 1 and SE 2 and containing automatic pressure-reducing valves. These valves are set to open at service water loop pressures of between 85 and 93 psi. (3)

Emergency water supply either to or from the neighboring reactor is available through the 24-inch high-pressure crosstie (see Figures 1, 8). One connection is made to the crosstie through a line connected to the intertie between process headers of pump sets 3 and 4. This line is installed principally for use when the reactor is down and providing backup to the neighboring reactor; in which case V-70 and HPC would be open.

The principal connections of the high pressure crosstie are made to the AB risers of both reactors, the connection being in the 105 valve pit. An orificed check valve is arranged at each end of the line, and each of these valves is by-passed by valve V-73, which is an automatic, quick-opening valve (air operated). A drop of riser pressure to a specified level will cause the valve to open and admit emergency process water either from the AB risers of the neighboring reactor (through the orificed check valve) or from its 165 valve pit. The trip-settings of V-73 are established according to the mode of operation of the reactor as follows:

TABLE 2
TRIP SETTINGS OF V 73 VALVE (4)

<u>Trip</u>	<u>Min. TORP*</u>	<u>Operating Condition</u>
1. Very low pressure	45 psi	Reactor operating
2. Extremely low pressure	20 psi	Reactor down, cross-under valve closed.
3. Extremely low pressure	4 psi	Reactor down, cross-under valve open.

*Top of Riser Pressure

In cases 1 and 2, the trip opens V-73 only, while the ELP trip for case 3 opens V-73 and closes the 36-inch cross-under valve if 4.16 KV power is still available.

The limiting orifice just upstream from each check valve originally was sized to pass 35,000 gpm with the neighboring reactor operating, and has a diameter of 10.6 inches. A measuring orifice is installed in the high pressure crosstie line at the location shown in Figure 1, to provide flow data during operation of the crosstie. Additional valves associated with the high pressure crosstie are the V-72 guard valves, V-71 flush valves and the V 74 drain valve (KW only). The V-72 is set to close by pressure switch if, during reactor operation, the crosstie pressure drops to a value of 30 ± 5 psig. All valves in the line are lubricated plug valves and all are motor-operated except V-73, which is air-operated. Operating times of the valves are as follows in Table 3.

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TABLE 3

VALVE OPERATING TIMES

<u>Valve</u>	<u>Time - Seconds</u>
All FB and PD valves; HPC	180
V-72	60
V-73 (full opening)	12 minimum ⁽⁴⁾
V-73 (20° to 70° opening)	1-4 (approximately)
V-73 closing	30 minimum ⁽⁴⁾

4.2 EMERGENCY GENERATOR SYSTEM

The primary source of power for the water plant is the 230 KV distribution system originating at the BPA midway substation. An additional (or alternate) source of power is made available through three steam-driven turbo-generator sets which provide 4.16 KV power to buses supplying selected pump motors and miscellaneous critical electrical components of the system. A detailed description of the electrical system is given by Robbins in his parallel report on the reliability of the electrical power and control system.⁽⁵⁾ This discussion will be limited to generator drive equipment.

Each generator is driven by a separate boiler and stream turbine. The boilers are oil-fired and are of the water-tube type. Each of the boilers has a separate fuel and feedwater system and normally supplies steam to its associated turbine. Principal auxiliaries, such as the feedwater pumps, forced-draft fans and air compressors, are essentially in two sets, one steam and the other electrically powered. The fuel pumps are steam-driven. Normal feedwater is applied through a deaerating feedwater heater, which draws water from the condensers, and makeup water from softeners (supplied from the service water loop). The main feedwater supply is automatically controlled by a steam flow rate meter.

Several alternate feedwater supplies are available in case of failure or inadequacy of the normal system, and these are enumerated as follows:

- a. A line from the service water loop connects to the feedwater heater and supplies heated, unsoftened water to the feedwater pumps.
- b. An auxiliary feedwater line runs from the discharge of each feedwater pump (there are two electric pumps and one steam pump per boiler) to a header leading to the associated boiler.
- c. Additional emergency supplies are provided from the service water loop and the condenser circulating pumps in the event of failure of the deaerating feedwater heater. These pipes lead directly to the feedwater pump suction lines.

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Steam lines from the boilers are connected to the associated turbines and to export lines providing low pressure steam for miscellaneous services within the entire plant. An automatic valve is arranged to shut off all export steam in emergencies where the generators are required to supply all power requirements of the system (BPA failure) at a steam load requirement of 53,000 lb/hr. at 400 psi. The steam piping is arranged so that the boilers and turbines may be interconnected in any desired manner.

The turbo-generator units are rated at 5000 KVA (3500 KW at 70 percent power factor) and two are normally operated in parallel with the BPA power supply. In case of separation from BPA, each of the two generators is arranged to supply a separate 4.16 KV bus which is loaded only with essential equipment. The third generator is carried as a maintenance spare and normally is not operated with the other two.

4.3 WATER PLANT CAPABILITIES AND MODES OF OPERATION

In order to facilitate discussion of the reliability of the water plant, it is considered necessary to include a brief summary of water flow requirements under various conditions, capabilities of the system, and a description of the various modes of water plant operation. These are discussed in the following paragraphs.

Water Requirements and Modes of Operation

According to the present operation of the water plant, either five or six pump sets are used to supply process water to the reactor at given power levels, depending upon the prevailing inlet water temperatures. Since the pressure-flow characteristics of the reactor depend upon the manner of its loading, the actual operating point for pump set discharge will also depend upon the reactor loading. According to the present (KIII) loading, operation of the pump sets is as shown in Figure 6. The indicated intersections of pump and reactor characteristics establish the normal process water flow capabilities. It should be noted that the pump characteristics as shown in Figure 6 have been corrected for piping system characteristics and static head differences.

The service water demands both during normal reactor operation and the minimum requirements during shutdown are shown in Table 4, below:

TABLE 4
SERVICE WATER REQUIREMENTS (APPROXIMATE)⁽⁶⁾

<u>Item</u>	<u>Normal Flow - GPM</u>	<u>Minimum Flow - GPM</u>
183 Filter Plant	up to 3400	0
190 Process Pump Cooling	3000	3000
165 Power Plant	300	300
105 Services	<u>3000-6000</u>	<u>3000</u>
	Max. 12,700	Min. 6,300

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Typical modes of operation of the KW water plant are shown in Figures 9-14. Although these diagrams were prepared for an analysis to determine the possibility of critical valving situations, they also are valuable in providing quick reference for valve settings and flow paths under the various conditions of operation shown.

In the diagrams, flow paths are shown by darkened lines, while closed valves are darkened. All valve positions shown were determined from the Power Operations' Operating Manual⁽⁷⁾ and/or a recent conference with a K plant shift supervisor.⁽³⁾

The normal modes of operation of the water plant are specified to a certain extent in Process Standards^(4,8). Generally, it is required that the low pressure crosstie be in service at all times and that a minimum two service water pumps are operated. A third service water pump is placed in operation when a reactor is shut down. It is specified further that two emergency generators must be in service at all times in both plants. Upon loss of one or two generators at a given plant, the associated reactor is required to shut down immediately at the normal rate, while the neighboring reactor is allowed to operate for a maximum of two hours if normal backup cannot be supplied by the down reactor within that period of time. Other requirements are illustrated diagrammatically in Figures 9-14. Alternate modes of operation will be discussed in the following brief summary of the charts.

Figure 9 - Normal 6-Pump Operation

In this case FB-5 may be open. In 5-pump operation, FB-5 must be closed unless No. 5 and 6 process pump sets are operating. The combinations of pump sets are optional in 5-pump operation. V-73 is set on "automatic" at a VLP trip of 45 psi TORP.

Figure 10 - Reactor Down

KW is down and is providing normal HPC backup for KE which is operating. In this case, low-lift pump No. 3 is valved directly to the high pressure crosstie through the valve HPC. If the backup is provided by pump No. 4, FB-5 must be open and FB-1, 2, 3, 4 must be closed. An alternate way to provide backup by a down reactor is to pressurize the high pressure crosstie through the V-72 valve from the AB risers of the down reactor. In this case a minimum of 5 low-lift pumps and 2 service water pumps are operated.

Figure 11

KW is down and receiving backup from KE (typical). Note that the low pressure crosstie is pressurized by three service water pumps.

Figure 12

KW is down and KE is flushing the high pressure crosstie through V-71 in KW. V-73 is closed and on manual setting, and V-72 is closed.

Figure 13

KW is flushing the high pressure crosstie through V-71 in KE. V-73 in KW is closed and on manual setting.

Figure 14

KW has the 36-inch cone valve in the cross-under open (for charge-discharge operations). HPC backup is provided to KE.

Process Water System Capabilities

Normal flow capabilities of the process pumps are as shown in Figure 6. It will be noted that no damaging cavitation exists for either 5- or 6-pump operation. No information appears to be available at the present as to the cut-off characteristics of the pumps to be modified under the K expansion program.

An important consideration is the performance of four pump sets at any of the reactor loadings shown. These pump sets operate well into the cavitation cut-off region, which means that, under these conditions, their flow capabilities are severely restricted. It should be noted further that flow increases beyond the present levels for 5-pump operation would place the pump dangerously near cut-off conditions.

The flow requirements and capability for the water plant during transient operation of a reactor caused by a scram are shown in Figure 15*. In this graph are plotted reactor water flow requirements for different power levels at summer operating conditions (20°C inlet, 90°C bulk outlet temperatures), a flywheel decay curve and various steady-state flow levels which correspond to the quantity of flow available during emergency flow conditions. These curves will be discussed in detail in the following paragraphs.

- a. Flywheel Decay Curve - This curve was derived from a composite of flywheel decay data obtained from records of tripout tests. All of the test data obtained corresponded to 5-pump operation, with initial flows ranging from 125,000 gpm to 154,000 gpm. It was found that when these curves were plotted on a percent initial flow versus time basis, all points fell very close to a single line. A conservative lower-limit envelope curve was determined, and this was plotted in Figure 15, with a time delay of 1.5 seconds to allow for rod drop in the reactor; thus the zero time origin for the flywheel curve is reactor time minimum 1.5 seconds. Incidentally the curve used corresponds almost exactly to one given in HW-42102.(9)

The discharge scale of the curve was adjusted to correspond to an initial reactor flow of 175,000 gpm (6-pump operation) which in turn approximately corresponds to a power level of 3400 MW (shown as a

*Flow requirement curves obtained from SS Jones (9-10-59).

broken line). The use of the 5-pump decay curve to represent 6-pump decay conditions is admittedly an approximation. It is, however, considered sufficiently accurate for this evaluation. Note that since reactor flow (steady state) is a function of power level, and since the percent initial flow concept of flywheel decay appears to be valid, instantaneous values of flywheel decay flow tend to rise with an increase in power level. This means that the decay curve as plotted in Figure 15 applies only to the 3400 MW level.

- b. Flow Capabilities Under Emergency Conditions - The flow levels indicated by the curves labeled A through C represent flow capabilities of the water system under failure of the 230 KV power supply (BPA). These flow levels were determined in a study of various documents concerning tests and/or computations of the capabilities of the water plant under conditions of BPA failure (10, 11, 12, 13), as discussed below.
1. Both reactors operating prior to BPA failure. Upon BPA interruption, two generators fail in one reactor and flow is carried to the stricken reactor through high-pressure crosstie from the A-B risers of the aiding reactor. The upper line represents flow to the aiding reactor, while the lower one gives flow to the stricken reactor. It will be noted that the latter falls below the 3400 MW flow requirement curve at about 150 seconds after the scram, indicating a deficiency in supply for a period of almost 250 seconds.
 2. One reactor supplying backup protection to the other prior to BPA failure (see Figure 10). Upon separation of the 13.8 KV bus, both generators in the operating reactor fail, and flow is made available to the stricken reactor through the HPCT from the 165 valve pit of the aiding water plant. As above, the upper level represents flow to the aiding reactor and the lower level the flow to the stricken reactor. Flow appears to be adequate for these cases.
 3. The initial condition could be interpreted the same as above, except that in the down area only two primary pumps are operating into the CD risers and FB-5 is closed. This condition could result from failure of one generator in the aiding area on BPA separation, prior to which this reactor was operating five process pump sets Nos. 1, 2, 3, 5 and 6 with FB-5 closed. Flow in this case is carried through the low pressure crosstie only. A flow deficiency exists for this case from a time of about 170 seconds after the scram to 2200 seconds.

The capability of the crosstie in providing safe shutdown flow under the foregoing conditions is highly questionable, since flow deficiencies exist in two cases, both near the end of the flywheel decay curve. Possible remedial actions for these deficiencies will be discussed in a later section of this report.

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4.4 BASIS OF RELIABILITY STUDY

In this study three different approaches were taken to assess the reliability of the K Water Plant. These are as follows:

- a. To evaluate the inherent reliability of major equipment or cooling system components to perform their design functions and to examine the operational and safety aspects of potential failures.
- b. To study the safety aspects of operation and control of the water plant.
- c. To determine the degree to which the design and capabilities of the water plant conform to safety criteria heretofore promulgated. These criteria, as presented in a recent document dealing with K Water Plant Improvements,⁽¹⁴⁾ are reproduced in their entirety in Appendix E.

The first of the foregoing approaches to the reliability analysis is very broad in scope and it was necessary to limit the reliability study of equipment and other plant components to determination of possible critical situations under various operating conditions. In the study of both equipment reliability and consequences of equipment failure, no conclusions are drawn relating to numerical probabilities of failure. The analysis dealing with the foregoing aspects of reliability will be presented in the section of the report dealing with a study of potential hazards.

The second approach to reliability, namely study of the safety aspects of operation and control, is also very broad in scope and time was not available to pursue this study to the extent that it deserves. Comments on the results of this study will also be included in a later section.

Classification of Cooling Systems

An analysis of the reliability of the K-Reactor Water Plant, in terms of the foregoing safety criteria, requires classification of the cooling system under the categories of these criteria. In this study, the classification was made as follows:

a. Primary Cooling System

This system includes the process pump sets, their associated piping system and valves, the 13.8 KV electrical distribution and control system, and all mechanical and electrical components which are directly associated with provision of coolant during normal reactor operation. As scoped in this report, the system is interpreted to include the portion of the water plant between the clearwells and the front crossheaders of the reactor, and is enclosed, essentially, in the 165 and 190 buildings.

b. Secondary Cooling System

Most of the hydraulic and mechanical components of the secondary system are common to both the primary and secondary systems, the principal difference being in the source of electrical energy supplied to the pumping system, which in this case comprises the low-lift pumps

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and the pipes which bypass the high-lift pumps. As previously described (Section 4.2), this power is supplied through the 4.16 KV emergency buses which are energized by the emergency generators. Thus the secondary system includes the boilers, turbo-generator sets, and their auxiliaries. Since a satisfactory transition from primary to secondary system operation depends upon capabilities of the flywheels of the high-lift pumps, these flywheels are considered to be integral units of the secondary system.

c. Last-Ditch System

The principal components of this system comprise the high-pressure crosstie, the low-pressure crosstie, and all associated valves and controls. Since these crossties of a given water plant are supplied by the primary or secondary systems of the neighboring reactor, both of these systems must be considered as being integral to the last-ditch cooling system of the subject water plant. The process pump set flywheels of the subject water plant are also considered as being components of its last-ditch system.

4.5 ANALYSIS OF POTENTIAL HAZARDS

In previous documents, much has been written on the subject of potential hazards and the analysis of failure of various water plant components. The reader is specifically referred to these documents for a complete background on this subject. (14, 15, 16, 17, 18) HW-59696(14) has the most recent summary of reactor hazards.

The study included in this report will be integrated with these previous studies and pertinent conclusions will be made in relation to the overall aspects of reliability and safety.

4.5.1 COOLING WATER SUPPLY

The clearwell supply of filtered cooling water would be considered entirely adequate if it were able to provide, within a reasonable margin of safety, a sufficient quantity of water to enable safe shutdown of the reactor upon loss of river pumps or other failure in the clearwell supply. In the design scope of the filter plant (19) the size of the clearwell was established to provide sufficient water for 30 minutes of operation at a flow of 140,000 gpm (maximum in initial scope), flow during designed shutdown condition for the next five hours and design shutdown flow of 3000 gpm for 24 hours. The capacity also was scoped to be adequate for service water demands during this period of time. The capacity based upon these demands was set at a minimum of 18,000,000 gallons of filtered water.

The minimum reserve for storage as specified in Process Standards (8) is based upon the foregoing water demands, with the requirement that an adjustment be made for the first 30 minutes of full normal flow according to the prevailing reactor water requirements.

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The results of two analyses^(1, 20) have indicated that the clearwell is very nearly adequate as scoped, and probably adequate under any conceivable situation. An approximate analysis was made by the writer on the basis of assumptions made in HW-39819⁽¹⁾ for the worst case of operation in which both KE and KW reactors were supplied by the clearwells of one plant only. This study involved graphical integration of the summer 4000 MW water requirement curve (see Figure 15) and was made on the additional assumptions that (a) the maximum flow of 21,600 gpm was being supplied to the neighboring reactor (while required) through the high pressure cross-tie;⁽¹³⁾ (b) the minimum shut-down flow was 5000 gpm for each reactor;⁽²⁾ and (c) reactor and thermal shield cooling, plus other miscellaneous demands was 3000 gpm for each reactor.⁽²⁰⁾

The approximate minimum time to empty the clearwell under these assumptions was found to be 17.5 hours, providing that the reactor was shut down (scrammed) immediately upon loss of river pumps. It is obvious that if the reactor shut down at the normal rate, the time would be somewhat less. A safety factor exists, however, in that an additional quantity of stored unfiltered water is available in the settling basins of the filter plant.⁽¹⁾ The possibility appears to be extremely remote that at least one river pump could not be restored to service within the necessary time. Providing that the water plants are operated in such a way as to conserve filtered water after all river pumps are lost, the storage appears to be adequate.

It is considered appropriate to comment also upon the emergency raw water supply to service water, process and condenser systems. It has been determined that, for all normal conditions of operation, at least one, and in many other cases, two valves in series leading from the raw water mains are closed. In the case of the raw water connection to the service water loop (low pressure crosstie) two series valves are closed for all conditions. It was found, further, that both of these valves must be manually operated. The manner of operation of these valves seems reasonable, in the light of possible consequences of contamination of the process or service water. If this philosophy is to be maintained, however, it would seem that the value of the raw water inter-ties as quick sources of emergency water would be lost. The operating time of the motor operated valves is three minutes; estimates of the time required to open manually operated valves range from five to thirty minutes.⁽³⁾ It should also be noted that the butterfly valves in the 60-in. raw water lines (RW1 and RW2) are operated from 4160 volt power, total power loss in a water plant would render these valves inoperable for practical purposes.

Process Standards⁽⁸⁾ states that the raw water strainers in the raw water inter-tie (between valves RE and RCE) should be checked at least once per year and cleaned as necessary to maintain low pressure drop. According to a shift supervisor,⁽³⁾ more frequent checks

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should be made and, possibly, some provision should be made for continuous flush during times of high raw water turbidity. This is particularly necessary because of the dependence upon the raw water system as an emergency source of condenser cooling water.⁽²⁹⁾ Each condenser in the power plant requires approximately 5000 gpm at maximum steam load⁽⁷⁾ and this would seriously deplete clearwell storage in case of total river pump failure.

No mention was found in Process Standards⁽⁸⁾ as to inspection and maintenance of the screens over the sluice openings leading from the clearwells to the 190 pumpwells. Progressive plugging of these screens (10 x 10 mesh to the inch) could cause excessive head loss between clearwell and pumpwell and this would possibly have a deleterious effect upon pump performance. Low pumpwell level alarms in the 165 control room remove the possibility of a hazardous condition resulting from the screen plugging. However, unexpected low level due to this effect possibly would cause unscheduled down time and consequent production loss.

4.5.2 RELIABILITY OF PRIMARY COOLING SYSTEM - PUMPING EQUIPMENT

The worst case of potential failure of the pumping equipment is considered to be that type of failure which would cause the pump set to lose the flow-decay capabilities of its flywheels. This effect conceivably could be caused by at least three factors, namely a shaft failure between the impeller and flywheel, rupture of the pump casing, or loss of one blade causing loss of the rest due to unbalance conditions. The first of these possibilities appears to be entirely credible, as evidenced by the cracking of the low lift pump shafts.⁽²¹⁾ Cracks and actual breaks in impeller blades are credible in open impeller pumps such as the low lift units. While a carefully planned and conducted inspection program will considerably lower the probability of such failures, it is assumed that the failures still could occur. Consequently, a brief study was made of the effects of pump failure of this type and the results are included in the following paragraphs.

For clarity, this discussion will deal with the effects of the loss of either the low-lift or high-lift unit in Process Pump Set No. 5, with all six pump sets operating prior to the occurrence of the incident, although it applies to many other combinations of pump sets. (See Figure 9)

The immediate effect of loss of either of these pumps would be a rapid drop in pressure in the 24-inch pump discharge line, the initial decrease being approximately equal to the pressure supplied by the stricken unit. In the case of low-lift pump failure, the pressure decrease would be rapidly augmented by added losses in the suction line of the associated high-lift unit and possible operation of this unit in the cut-off region due to cavitation.

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Because of the high velocity of pressure wave propagation in the 24-inch pipe (approximately 4300 ft/sec) and the short lengths of piping involved, it is assumed that the pressure decreases would be felt almost instantaneously throughout the piping of the process water system. The surging action which could take place could subject much of the system to severe dynamic stresses, particularly in the case of failure of a high-lift unit, which would produce an initial pressure decrease of about 400 psi.

A simplified (and possibly conservative) approach to calculation of the magnitude of pressure surges can be made, assuming (among other things) that the process piping is continuous between Process Pump Sets No. 4 and 5. This is reasonable since it can be shown that the velocity change taking place across a pressure wave is not sufficiently great to cause reversal of flow in the piping; consequently the check valves will remain open during this phase. If the foregoing assumption is made, it will be seen that the negative pressure wave originating at the stricken pump set will return to this point as a negative surge with the magnitude of double the initial surge. The actual conditions are complicated by the following considerations:

- a. wave reflections from complex pipe junctions and bends,
- b. wave reflections through piping connected to the reactor and to other pumps, and
- c. speed of pressure decay at the discharge of the affected pump set, and stability of remaining pump sets.

Because of these considerations, little confidence can be placed in such a simplified analysis. The problem should receive further analysis, considering in detail the effects of the foregoing items.

The rate at which pipe flow decreases occurred would be somewhat limited by inertial effects of water in the large pipes, but would be rapid because of the potential speed with which the pressure decrease could take place. It is important to note that, since pump speed would be constant during the transient, the flywheels of the units remaining in operation could not reduce the rate of flow decay caused by failure of a single unit with loss of flywheel action.

The failure of the low-lift unit of a pump set would not result in reversal of flow through the associated high-lift unit, since the shut-off head of the high-lift pump is greater than the operating head of the remaining pump sets. (See Figures 2, 3 and 6) However, if a high-lift unit failed, a flow reversal could take place in the low-lift unit and in the 24-inch discharge pipe, and this could lead to serious damage to the system due to rapid closure (slamming action) of the check valve near the pump discharge valve. This results from the fact that the line from Pump Set No. 5 would be pressurized by Pump Set No. 4 through valve FB-3 (Fig. 9). Rapid closure of the

check valve could result in serious water-hammer effects in the process piping and, possibly, failure of the check valve itself. This could result in a serious reduction in flow of coolant to the reactor.⁽²²⁾ A study of Figure 9 will show that other check valves (FB and high-lift pump bypass) in the process piping could be subjected to the same action.

The possibilities of damage caused by flow reversal are not peculiar to the process piping alone. If the flow-balancing valve FB-5 were closed, sudden reduction of pressure on one side of the water plant would cause flow reversal to occur in the front crossheaders of the reactor, which could cause the check valves at the low pressure side to close very rapidly (risers A&C and B&D are connected by the cross-headers). Again, there exists the possibility of pipe rupture or valve failure.

As with pressure surges, the seriousness of effects of flow reversals would be dependent upon the rate of pressure decay initially occurring on pump failure, and the inertial effects of water in the process piping.

The flow capabilities of the process system would be reduced by approximately eight percent for initial 6-pump operation and 19 percent for initial 5-pump operation, after abatement of the initial surges. This assumes no flow contribution from the stricken pump set and that no piping or valve failure occurs. In both cases there is a strong possibility that flow conditions would be unstable, particularly for the case of initial 5-pump operation. From Figure 6 it is evident that the four remaining pumps would possibly operate in the cut-off region and thus would be subjected to severe vibration conditions.

Consequences of Failure

The consequences of the various types of failure discussed in the foregoing paragraphs cannot be assessed at the present time because of the many complications of the problem. It is evident, however, that they range from momentary sudden reduction of flow to almost complete loss of flow to the reactor. Some of the variables which must be considered in a detailed analysis of the many facets are as follows:

- a. Rate of pressure decrease caused by the pump failure with loss of flywheel action.
- b. Inertial effects of the water in the discharge piping.
- c. Speed with which the safety circuits or panellit guages cause the reactor to scram.
- d. Transient characteristics of the pumps under the different situations discussed.

- e. Vibration characteristics of the discharge piping.
- f. Ability of the check valves to withstand sudden pressure changes.
- g. Combinations of pumps operating and valving setup (particularly FB-5) prior to the incident.

The consequences of pump failure without loss of flywheel action do not appear to be serious in the light of one and two pump trip-out tests reported by Jones.⁽²³⁾ In this case a gradual flow transition between 6- and 4-, and 5- and 4-pump operation is provided by the flywheel action of the tripped-out pump sets. Another case, which most nearly approximates the condition studied, was in the failure of the emergency generators to assume their load in the incident of August 21, 1955.⁽²⁴⁾ In this case the 4.16 KV bus was separated from BPA and the 13.8 KV system remained operative. The result was that the low-lift pumps were tripped out and the high-lift pumps operated under severe cavitation conditions for 43 minutes. This is not exactly analogous to the cases presented in this report, since the flywheels of the low-lift pumps provided a more gradual flow decay in that situation.

Cavitation

Cavitation itself need not be considered a cause for failure of pumping equipment, providing that adequate maintenance is programmed to inspect impellers and casings for cavitation damage, and providing that the pump is not required to operate in the cut-off region. In this case, the most serious consequence, not considering the effect on associated equipment, is an unexpected reduction of pump capacity. An example is the possible cavitation in a high-lift pump during normal speed decay. Most cases of this type, where the cavitation cut-off and consequent flow reduction is suspected, have been analyzed in other studies.⁽²⁵⁾ Operation of the pump sets at extremely high discharges should be accompanied by careful checks for cut-off conditions and tests should be conducted, to determine the presence of these conditions, for all conceivable modes of pump operation in which cavitation is suspected.

4.5.3 RELIABILITY OF THE PRIMARY COOLING SYSTEM - PIPING AND VALVES

The reliability of the piping system is based essentially upon its ability to withstand vibrations (experienced in normal operation or caused by earthquake conditions), cavitation damage and sudden pressure changes caused by waterhammer. Although none of these cases were analyzed thoroughly in this study, it is pertinent to include some observations on each of the various factors.

- a. Vibrations caused by turbulence associated with high velocity flow. The consequences would appear to be in embrittlement fatigue failure of piping and/or welds.

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- b. Earthquake Vibrations - The existence of possible resonance conditions in various portions of the process piping should be studied, as recommended by Roberson in his study of possible failure consequences of pipe rupture.⁽²²⁾
- c. Cavitation Damage - There is very little chance that cavitation would occur in the piping leading from the discharge of the high-lift pumps.

A brief study of the suction piping at the vaned elbow preceeding the higher lift suction elbow based on model studies made by Behlke and Pritchett⁽²⁶⁾ indicated that little danger of cavitation existed at that critical location. In the model tests it was determined that no cavitation occurred at this point if the cavitation parameter σ^*

$$\sigma^* = \frac{NPSH}{v^2/2g} - 1$$

was greater than 6.0. Calculations by the writer based on low level (12.5 ft. depth) for high-lift pump No. 1 and a pump set flow of 40,000 gpm yielded a cavitation parameter of approximately 20. It should be noted, however, that cavitation may be caused by vibration.^(27, p. 237) Consequently careful inspection of certain suction piping may be required; particularly at points where boundary layer separation or low pressures may exist.

- d. Waterhammer - Potential causes of waterhammer include the effects discussed previously or cases in which valves are suddenly opened or closed. The V-73 valves in the high pressure crosstie and check valves are probably the only sources of sudden flow changes, because of the slow speed of other valves and the short lengths of most pipes. The worst consequences of waterhammer are bursting of a line, collapsing of a line or failure of welded connections. Pipes previously weakened by cavitation, corrosion or fatigue would be particularly subject to this type of damage.

The reliability of most valves in performance of their design functions is easily determined by testing. Critical valves such as V-73, V-72, and flow balance valves, which must operate under extreme conditions (such as line rupture), can be tested through the flush line of the high pressure crosstie. Functional tests of other valves, such as emergency service water valves (SE), process to service water valves, and check valves should be made at regular intervals. Some of the check valves such as those in the reactor crossheaders and those at the bottom of the A-B risers are considered critical from the stand-point of pipe rupture. Proper testing and maintenance should preclude any possibility of valve failure under design conditions of operation.

4.5.4. RELIABILITY OF THE SECONDARY COOLING SYSTEM

As defined, the secondary system is included to provide adequate cooling to the reactor in the event of a failure of the primary system, separation from BPA, or during normal slowdown conditions. Most of the cooling system components utilized in the secondary system are common to both this and the primary system. Consequently the only additional reliability considerations are related to operation of the emergency generators and certain valves.

The electrical aspects of the secondary system are discussed by Robbins in his parallel study of the water plant.⁽⁵⁾ This discussion will be limited to the steam equipment and certain components of the water system not treated of in the previous section.

Steam Plant

The steam plant generally is considered adequate to perform its design function, as demonstrated by the large number of successful acceleration tests run to date.⁽²¹⁾ In relation to reliability of critical auxiliaries, a detailed analysis revealed that this system was more than adequate to supply the boiler under full capacity conditions (58,000 lb. of steam per hour). In addition, it was discovered that in their acceptance tests, the boilers were successfully run at rated conditions (53,000 lb. per hour) for eight hours, after which they were operated at full capacity for two hours.⁽²⁸⁾

Several discussions were held with steam plant operators and shift supervisors in KE to determine their opinions on reliability of the steam plant. The general consensus was that the reliability was good, with the following exceptions:

- a. The fuel oil pump capacity is only barely adequate for full load boiler operation.
- b. An over-riding clutch is required on the turbine shaft of the dual-drive forced draft fan. One to two hours of boiler downtime is required for maintenance of the turbine (replacing packing glands and other small but important details). It is understood that an over-riding clutch is provided, but that this only permits the turbine to over-ride the electric drive.
- c. The water softeners in the boiler feed makeup system are adequate for full generator load conditions only when no steam is exported.

Piping System Components

Assuming either 5 or 6-pump operation prior to a scram of the reactor caused by BPA separation, the shut-down flow after flywheel decay is more than adequately provided by the pumps remaining on the 4.16 KV emergency buses. Failure of a single bypass check valve to open would

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create a hazard only under the circumstance occasioned by loss of one generator and simultaneous failure of three or four pumps, assuming 5 or 6-pump operation prior to the BPA loss. This case is considered very unlikely. It appears that proper functioning of the equipment prior to a scram under these circumstances, coupled with adequate testing and maintenance, would almost automatically insure proper functioning of the secondary cooling system piping and valves.

There remains the question of the adequacy of the secondary system in the event that a scram was caused by malfunctioning of certain mechanical components of the primary cooling system. If the scram were caused by failure of a pump set without loss of flywheel action it seems reasonable to assume that the secondary system would function properly after the flow decayed to a satisfactory level.

In the case of a scram caused by failure of a pump set with loss of flywheel action, it would appear that the secondary system still would be adequate after a short period of time (probably about five seconds if the scram was initiated by electric trip* and longer if panellit or riser pressure trips actuated the scram). If the pump set failure were accompanied by pipeline failure, the secondary system could also fail. This would be particularly true if valve FB-5 were open prior to the scram.⁽¹⁸⁾

Because of the considerably lower pressures and flows in the secondary piping system than those in the primary system, piping failure does not seem to be credible. Loss of a pump set, however, could cause failure of the system in the event a reactor was down and supplying backup to the neighboring reactor with one low-lift pump as shown in Figure 10, but with only one low-lift pump supplying the associated reactor. It would appear that the most simple approach to solution of this problem would be in specification that a minimum of two low-lift pumps be used to supply cooling water to a down reactor. If the down reactor is supplying backup for its neighbor, a minimum total of three low-lift pumps should be operating.

4.5.5 LAST DITCH SYSTEM

The high pressure crosstie (Figures 1 and 8) was designed to provide cooling water to a given reactor upon failure of its primary and secondary systems. The water is provided from the neighboring reactor through the V-73 quick-opening valve in the stricken reactor and either the orificed check valve or an open HPC valve of the neighboring reactor.

Several cases of the flow capabilities of the high pressure crosstie are shown in Figure 15, along with the corresponding conditions of combined reactor operation. The worst case, which represents flow

* Estimated from trip time of 2 sec.⁽²³⁾ and reactor water requirement curves.

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through the low pressure crosstie only, constitutes almost total failure of the high pressure crosstie. This could be occasioned by the following situations:

- a. Both plants operating prior to BPA separation. Plant A has FB-5 closed and pump sets 1, 2, 3, 5, and 6 operating. HPC and V-73 closed in both plants. BPA fails and causes both reactors to scram and, due to relay faults,⁽¹⁴⁾ three of the four generators in the two plants fail, leaving only pump sets 5 and 6 of reactor A operating. High pressure crosstie flow is not available to plant B.
- b. Plant A is down and providing backup to Plant B through the HPC valve as shown in Figure 10. A failure of pump set No. 3 and simultaneous failure of all power at Plant B would cause failure of the high pressure crosstie to supply flow to the stricken reactor.

Neither of these possibilities seems very credible since each one requires three simultaneous equipment failures.* It should be noted also that neither case represents total loss of water and that adequate service could be provided by starting a low-lift pump and opening FB-5 and PD-4. The valve opening time is about three minutes so that service probably could be restored within about five minutes. It may be possible that some damage to the reactor would result, but there seems to be little likelihood of a major disaster.

A conclusion that may be drawn from the foregoing discussion is that adequate supply of pressure and flow is available to the high pressure crosstie connections (at A-B risers or HPC line) under all cases if prevention of a major disaster is considered as the criterion for safety.** The probability of total failure of the pressure and flow supply appears to be negligible, and that of momentary failure seems to be very low. There now remains the question of reliability of the mechanical components of the system and these are discussed as follows:

- a. V-73 Valve - This air-operated valve is set to open automatically upon specified drop in riser pressure at the stricken reactor at the trip settings discussed in a previous section. According to a discussion with S. S. Jones,⁽²⁹⁾ the present level of performance of this valve is satisfactory, with the exception that the timing is very difficult to control. To avoid waterhammer effects in the crosstie piping it is important that the minimum safe times of opening or closing be carefully established and that provisions be made in the mechanism to prevent the valve from operating in any shorter times than these.

* There is a casual relationship between BPA failure and generator failures; so that this could be construed to represent two simultaneous failures.

** Except as noted on page 16.

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Failure of the V-73 valve to operate could be caused by loss of air pressure, mechanism failure, electrical failure of the trips or operator error in either failing to adjust the trip setting to the required level or leaving the valve control on manual setting. The consequence of failure of the valve to operate would constitute total failure of the last-ditch system, if the contribution of the low pressure crosstie is ignored.

Some analysis has been carried out to determine the effects of inadvertent opening of the V-73 valve in a down reactor upon flow through the operating neighboring reactor.⁽³⁰⁾ This analysis determined that the flow reduction could be up to seven percent if the reactor was operating six pump sets, and thirteen percent for 5-pump operation. It is expected that this flow decrease could be quite sudden because of the small amount of momentum present in the water in the reactor tubes. It should be noted, however, that the water in both the risers and process headers has a great deal of momentum and that in the crosstie piping has some inertia, all of which could limit the flow decrease that would occur. Tests are recommended to determine the transient which would occur under these circumstances.

- b. V-72 Valve - This valve is a guard valve provided to protect against the effects of rupture of the high pressure crosstie piping. Proper maintenance and routine testing should insure proper functioning of this valve.
- c. High Pressure Crosstie Piping - This piping does not appear to be as susceptible to vibration or waterhammer damage as is the process piping. Waterhammer effects could be experienced by too-rapid operation of the V-73 valve and the pressure surges should be studied for the case of most rapid opening of this valve. If provisions can be made for absolutely preventing the opening time from being less than that presently existing, then the problem can be solved by tests determining actual pressure surges and subsequent structural analysis of the effects of these surges. Both negative and positive surges should be considered.

In reviewing previous documents treating of the high pressure crosstie problems, some mention was found of difficulties occurring from corrosion of the piping.⁽¹²⁾ Specifically, the troubles were founded in the dislodging of particles of corrosion products from the pipe walls by high velocity flow, which particles plugged crossheader and venturi screens in the reactor. Since no mention of experiences with this trouble was found in any more recent document than that cited, it is assumed that the problem has been solved by more frequent flushing; if not, some remedial action should be taken.

4.5.6 RELIABILITY OF OPERATION AND CONTROL

The manner in which the reactor and its various components is operated and the controls which are provided to handle emergency situations, are extremely important from the standpoint of reactor safety. Weak links in this system, which include operator training, coordination of operation of all plant components, adequate provision of safety alarms and trips, etc., could render the entire system inadequate. For this reason, continuous evaluation and review of the operation and control aspects is absolutely necessary.

Insufficient time was available in this analysis for a detailed study of the problem; however some discussion will be presented later based on a cursory examination of the situation as evidenced by discussions with engineers, operating people, visits to the plant, etc. It appears that some hazards possibly exist due to the multiplicity of operation and the inter-dependence of reactors and their water plants.

4.6 EXTENT TO WHICH SAFETY CRITERIA ARE SATISFIED

In this section a discussion will be presented in relation to the degree to which the safety criteria as presented in Appendix E are satisfied. In the following section, general recommendations will be made as to what remedial action appears to be necessary to correct any deficiencies noted.

4.6.1 PRIMARY COOLING SYSTEM

This system appears to adhere to all safety criteria except one, and this applies to the consequences of failure of a single component. If failure of a pump with loss of flywheel action or inadvertent opening of a V-73 valve at a down reactor is considered credible, and if a more detailed analysis shows that this would result in damage to the reactor, then failure of one such system component would be considered to render the primary cooling system inadequate. The effect of such a failure could cause a sudden reduction in coolant flow through the reactor or an almost total loss of water (in case of a pipe break).

4.6.2 SECONDARY COOLING SYSTEM

This system fails to meet the safety criteria in at least two counts, both in the requirements for independence. These are as follows:

- a. Electrical tie between BPA and the emergency generators. The aspects of this are discussed in HW-59696.⁽¹⁴⁾
- b. Common pumping and piping system. If failure of a pump could cause a failure of process piping, there is a possibility that the result would be in rendering the secondary system inadequate,

essentially by failure of a component of the primary system. If conclusive proof is available to show that such a condition could not occur, then failure of a single pump would not render the secondary system inadequate. If the pump flywheels are considered as components of the secondary system, then failure of a pump with loss of flywheel action implies failure of the secondary system.

4.6.3 LAST-DITCH SYSTEM

This system does not meet the safety criteria for two reasons as follows:

- a. The system is not adequate in two cases shown in Figure 15, at least one of which (Case A) appears to be credible.
- b. The failure of one component of the system can render the system inadequate. If the V-73 valve fails to open, the system would be rendered inadequate until such a time as flow could be made available through the HPC and V-70 valves.

4.7 RECOMMENDATIONS FOR REMEDIAL ACTION

In the following paragraphs are presented a summary of remedial actions considered necessary in order to provide for operation of the K Water Plant within the doctrine as set forth in the safety criteria for reactor cooling system.

4.7.1 PRIMARY COOLING SYSTEM

The principal deficiency of the primary system is an apparent susceptibility to effects of sudden reductions of flow caused by pump failure, pipe rupture or inadvertent opening of the V-73 valve in the high pressure crosstie. It appears that action should be taken along several lines, but that the principal measure should be to provide some emergency reserve of high pressure water to the reactor during the time when a flow deficiency exists.

Of the many possibilities, three are listed, with brief comments where considered appropriate.

- a. Pressurized tank having a sufficient volume of stored water at riser pressure. One should be placed on each pair of risers, with check valves on all risers as are presently installed on the AB risers. This would suffice only for cases when the flow deficiency was shortlived, but probably would not be economical if provision was required for a pipe rupture. These tanks could also be used to supplement the high pressure crosstie during BPA scram conditions. The tank at the C-D risers would serve as protection against failure of check valves in the crossheaders and help equalize the flow to the process tubes.

- b. Adding a seventh pump set as a maintenance spare, and operating six pumps (with crossheader throttling) so that the flow deficiency caused by loss of one would have a negligible effect upon the reactor fuel elements and tubes.
- c. No addition of pumps, but operating the six pumps as above. At times when one pump is down for maintenance, this would require operation of five pumps at the safe level (i.e., no reactor damage caused by temporary flow deficiency). This action is recommended to be taken immediately and continued until other remedial measures can be taken.

To prevent serious reactor damage caused by pipe rupture or check valve failure on loss of a pump, it would be necessary to take measures to prevent dangerous pressure surges from occurring in the process headers. If necessary, damping of the pressure surge initially occurring when the pump fails would require a surge tank or chamber located very near the pumps (possibly connected to the piping between the low-lift and high-lift pumps). This would, in part, compensate for the loss of the flywheel action. Probably the most important consideration is provision of non-slam check valves in the pump discharge lines to protect the piping between the valve and the reactor. This would also prove to be beneficial to the rest of the piping.

If a detailed analysis showed that it would not be possible to prevent damage to the reactor by providing the foregoing measures, then it would be important to introduce a safety device which would immediately scram the reactor at the time the pump failure occurred. Such a device could be pressure actuated or possibly could be actuated by any sudden change in current drawn by the associated pump motor.

4.7.2 SECONDARY COOLING SYSTEM

The only apparent course of action, to provide a secondary system which is independent of the primary system, is to install a completely new system, because of the existence of casual relationships between the BPA system and the present emergency generator arrangement. Isolating the emergency generator buses from the 13.8 KV system would remove the casual relationship between primary and secondary systems, but in this case, because of the use of common pumping equipment by both primary and secondary systems, the primary system would be strongly dependent upon the secondary system from a hydraulic standpoint. Independence in this case could be achieved only by adding to the present process pumping system a sufficient number of pumps to satisfy the requirements of the secondary system (over and above primary system requirements).

If it is possible to preclude the danger of pipe rupture, then the steam-driven pump backup as proposed for Project CGI 844(32,33) seems suitable, whereby the turbine pump would draw its steam from the

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spare boiler in each plant. This scheme has merit, in that failure of a generator does not necessarily include failure of the associated boiler; thus there probably would be at least one additional boiler available to serve as a backup for the boiler supplying the steam pump. Because separation from BPA could cause failure of secondary cooling system relays or switchgear, the turbine-driven pump should be idling at all times against a check valve (possibly with an automatic bypass line which would close when the pump picked up the load).

If it can be shown that failure of process piping is possible, then the secondary system could be independent of the primary system only if separate piping were run from the steam driven pump to the risers of the reactor. This line would require valves located so that rupture of process headers could not cause it to fail.

4.7.3 LAST-DITCH SYSTEM

There are two basic requirements for improving the last-ditch system to meet the safety criteria as follows:

- a. The capacity must be increased to the level at which adequate flow of coolant will be supplied to the reactor in the event of complete loss of primary and secondary systems. The scope of this requirement is very broad, since the effects of loss of a pump and possibly those of line rupture must be considered in addition to the flow deficiency indicated in Figure 15.
- b. Either the V-73 valve must be made fail-safe or an alternate means of automatically providing water to the reactor should be installed (another valve in a bypass line or possibly an electric drive for the existing valve), or the valve should be abandoned in favor of separate pumped or stored supplies to each reactor. The latter scheme has merit in that it removes many of the present complexities centering on the various possible modes of operation of the two reactors, and the very important fact that, due to the 13.8 KV crosstie between the two reactors, there is a casual relationship between the two primary systems. Thus, in very broad terms, there is an inescapable dependence between primary cooling systems of both reactors and, due to further casual relationships, a dependence between secondary and last-ditch systems.

The requirements given in the first of the foregoing items would be very difficult to meet, if all the possible modes of failure are assumed to be credible. This would require a system, having almost the full capabilities of the primary system, to provide adequate cooling in case of pipeline rupture. The only solution appears to be that of modifying the primary system and operating it in the manner previously discussed, such that the effect of failure of the primary system would not render the capabilities of the remaining flywheels inadequate.

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The required capacity increases for the last-ditch system probably can best be achieved through abandonment of the high-pressure crosstie in favor of the systems recommended in HW-61045 (Re: CGI-844), or one of the following schemes.

- a. Large reservoir of stored water on Gable Butte or Gable Mountain⁽³¹⁾ to supply backup for all reactors through the existing export lines and a line running from the K reactors to the export lines.
- b. High tanks to provide flow during the present periods of deficiency (Figure 15) or to supplement the pumped supply proposed for CGI-844. Some consideration should be given to the development of a temperature controlled valve leading from a pressurized tank, which valve would automatically admit cooling water to the risers upon increase of bulk outlet temperature beyond a certain specified limit. This arrangement, which would supplement the other systems, would be a last-ditch system in the true sense of the word, in that it would provide for inadequacies of any of the other systems.

If the present high pressure crosstie is not to be abandoned, several modifications should be made in order to improve its reliability, and these are as follows:

- a. The V-73 valve operator should be modified so that its opening time (particularly between the 22° and 70° travel interval) may be regulated to a greater degree of precision. Further recommendations, based upon suggestions by S. S. Jones,⁽³⁴⁾ are that a constant speed electric drive be installed to improve regulation of opening and closing time (this would require air backup), and that the VLP trip setting be increased in order to decrease the pressure drop across the valve during the time it opened. The latter modification would tend to reduce the pressure surges occurring in the crosstie line during valve opening or closing.
- b. Providing an interlock on the manual opening circuit of the V-73 valve such that the valve could not be opened manually at one reactor unless a corresponding action controlling the same valve was made in the other reactor. An alarm system should be provided between reactor control rooms to signal necessity for emergency manual opening of the valve.
- c. An additional pipe from the high-pressure crosstie should be run to the other side of the FB-5 valve in the tie line between No. 3 and 4 process headers, the HPC valve should be relocated in order to serve both connections, and check valves should be installed, allowing flow to the other reactor only. This would remove the present difficulties occasioned by operation of the FB-5 valve.

- d. The restrictions should be removed from the high-pressure crosstie line. This modification is only possible when high tanks (or pressurized tanks) are installed at the risers in order to compensate for momentary flow deficiencies caused by the opening of a V-73 valve.

Some mention has been made in previous studies of the use of raw water as a coolant source for the last-ditch system. This is not considered feasible for the following reasons:

- a. According to sanitary code standards, check valves are not adequate to prevent contamination of one system by another. Standard practice dictates that two closed valves be placed in series and this is a condition which would detract from the effectiveness of such a system.
- b. Strainers must be provided to prevent plugging of the fine screens in the reactor components. Such strainers are difficult to maintain. Sediment tends to collect in the piping unless water is continuously recirculated and corrosion would be difficult to control.
- c. Great difficulties exist in provision of arrangements for routine in-service testing of the system components.

Unless these (and probably many more) problems can be solved, it is recommended that any thoughts, regarding use of raw water for this purpose, be abandoned.

4.8 GENERAL RECOMMENDATIONS

This section is divided into four parts, dealing with recommendations on problems requiring further study, maintenance and testing, modes of operation and reactor plant control. These will be discussed in the following paragraphs in the order given.

PROBLEMS REQUIRING FURTHER ANALYSIS

Because of the large scope of the problem involving water plant reliability, it was impossible to make detailed solutions of all problems in the time allotted for the study. In addition, at least some of the problems are sufficiently complex as to require study by specialists in certain fields. Undoubtedly, there are many more areas in which detailed study is required, but analysis of the problems listed below is considered paramount to increasing reactor reliability and safety to the required levels.

- a. Further analysis of the pressure and flow transients occurring in process headers caused by loss of a pump, with or without flywheel action.
- b. Analysis of waterhammer caused by the loss of a pump without flywheel action. This study also should include an investigation into the structural integrity of both ordinary swing, and non-slam check valves.

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- c. Study of the stability characteristics of the system containing four or five operating pump sets and a single primary or secondary pump, operating in conjunction with an inoperative unit.
- d. Study of the waterhammer effects caused by too-rapid opening or closing of the V-73 valve, or the V-72 valve. This analysis should be followed by tests, if feasible, to determine actual stresses caused by the waterhammer.
- e. Investigation of the vibration and cavitation characteristics of high velocity flow in pipes and pumping equipment. This study should be accompanied by an examination of the metallurgical aspects of this vibration as applied to both piping and welded connections. Also included should be development of some means to detect cavitation in a qualitative fashion so that correlation can be made between degree of cavitation and the damage it produces. A possible means of accomplishing this is through oscilloscope studies of noise level peak intensities. Radiograph studies of piping and welds, to detect potential failures due to vibration effects, are considered essential to the integrity of the system.
- f. A detailed study and re-evaluation of the emergency alarm and trip systems should be made by an independent specialist in this field. A careful correlation should be made between equipment defects and their potential effects upon reactor safety. A study programmed for solution by an IBM computer possibly would reveal the presence of certain critical system components (this is particularly true of complex systems such as the emergency generators and their prime movers).
- g. The adequacy of the capacity of the clearwells should be constantly re-evaluated in terms of system flow demands as power levels increased. Study should be devoted to determine the feasibility of providing a tunnel to tie together the clearwells of the two plants.

RECOMMENDATIONS FOR EQUIPMENT TESTS

Certain problems in regard to water plant reliability can be solved by actual in-service or simulated-service tests of equipment or components. The writer is aware of only some of the routine tests conducted at present and some appearing in the below list may already be incorporated into the operating requirements of the water plant. It is important to emphasize that not only do periodic functional tests provide valuable data on equipment reliability, but also insure that a piece of equipment is operated, and frequently enough so that the maintenance problems can be discovered. Such tests also tend to familiarize operators with the mechanical equipment and other components of the complex system. Tests which actually result in safety hazards should be avoided; however those which result in normal functioning of reactor safety mechanisms due to equipment failure serve the valuable purpose of demonstrating the need for remedial action.

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- a. Emergency generators - The present tests of this equipment involve acceleration tests, conducted during scheduled reactor outages. The safety criteria doctrine, requiring that the secondary cooling system shall provide adequate flow indefinitely, requires knowledge of the ability of this equipment not only to accelerate to the required load, but also the ability of maintaining this load for extended periods. Thus it is recommended that periodic tests be made under rated load for long periods of time, possibly up to 24 hours. These tests would help to determine reliability of all components of the system under more realistic conditions.
- b. Pump maintenance - Because of the potential hazard associated with loss of a pump set, maintenance and inspection of the process pumps (particularly the low-lift units) is extremely important. The inspection should be carried out with sufficient frequency to enable detection of clues to incipient failure of shafts, pump casings or pump impellers.
- c. Cavitation - Tests for existence of both damaging effects and flow-limiting effects of cavitation should be made for all normal conditions of both steady-state and transient operation of pump sets or individual pumps. This has been done to a great extent with the existing pumps under previous conditions of operation. The testing necessarily should be kept in step with reactor flow increases or changes in mode of water plant operation.
- d. Waterhammer in the high pressure crosstie - As previously mentioned, tests should be made for either over- or under-pressure effects of opening and closing the V-73 valve. These tests should be preceded by an analysis to determine safe limits for the waterhammer tests.
- e. Experiments should be conducted to determine the actual pressure and flow transients occurring due to inadvertent opening of the V-73 valve under conditions such as are described in HW-61617.(30)
- f. Valve tests - Functional tests of all valves in the system should be made at regular intervals when possible. The valves singled out as particularly important are as follows:
 1. All check valves, particularly those in the pump bypass line, process headers and in the crossheaders on the front face of the reactor.
 2. All automatic valves, such as V-72, V-73, process to service water (165 valve pit) in the boiler feedwater lines of the steam plant, and relief valves.
 3. Flow balance valves, particularly FB-5. This and other valves which must operate in case of a pipe rupture, should be closed under the most stringent conditions it is possible to devise.
- g. Strainers and Screens - Rigid maintenance and inspection should be made of all strainers and screens in the system at regular intervals.

- h. Alarms and Trips - In many cases, normal or transition operation will indicate the reliability of safety alarms and trips, and checks of these should be made and entered in the operators' logbooks in every case an alarm operates. Some alarms, such as those indicating temperatures, fuel tank levels, etc., may never or seldom be required to serve their functions during the life of the plant. These alarms should not only be tested by routine continuity checks, but also by injection of spurious signals at the source, and this should be done at regular intervals.

MODES OF WATER PLANT OPERATION

The analysis of potential hazards and indicated certain requirements for safe and reliable operation of the water plant. These are summarized below:

- a. A down reactor should be supplied cooling water by at least two low-lift pumps, valved in such a manner that failure of one will not cause failure of the other. A minimum of three low-lift pumps should be operated if the down plant is supplying backup protection to the other reactor through the HPC valve.
- b. The present policy, requiring a minimum of two emergency generators to serve a water plant while the reactor is operating, is sound.
- c. As long as the possibility of pipe rupture is considered credible, the flow balancing valve FB-5 should be kept closed as presently specified.(8)
- d. Six pump sets should be operated in such a manner that effects of sudden deficiencies of flow which might occur as a result of pump failure or inadvertent operating of the V-73 valve would not cause damage to the reactor. When five pump sets are operated, the crossheaders should be throttled to the point where, if one pump were lost, the remaining four would not go into cavitation cut-off conditions.
- e. Under total BPA failure conditions, a chart should be provided the water plant operator which specifically establishes the manner of operation required to conserve clearwell storage of filtered water. There would be a great advantage provided in the installation of level alarms which would progressively warn the operator of the rate of pumpwell level decrease.

GENERAL CONTROL AND OPERATION

Some observations are considered pertinent concerning the important aspect of overall operation of the K reactors and their water plants. The first of these observations was not the result of a detailed study and is included here only as a suggestion aimed towards improving the overall reliability of the system from the control standpoint.

- a. Central Control - A prominent defect in the overall operation of the reactors and their water plants is in multiplicity of control arising from the complexity of the system. In cases of certain equipment failures or other emergencies, it is possible that the responsible operator or shift supervisor would not be fully aware of the possible consequences and would fail to take the necessary measures until it was too late. In short, decisions in certain cases must be made by several men before appropriate action can be initiated.

For example, let us consider a case in which an alarm associated with a high-lift pump indicates some trouble, such as high pump bearing temperature, high gear bearing temperature, low cooling water pressure, etc. Decisions must be made on various levels as to how serious the trouble is with respect to the pump itself, to the water plant, and to both the associated reactor and its neighbor. Personnel involved in these decisions would be the 190 pump operator, the 165 control room operator, the water plant shift supervisor and operators and shift supervisors of both reactors. These intermediate decisions would not be made simultaneously, but would probably follow the approximate order of the various levels listed above. It is conceivable that, in spite of the excellent communication facilities available, considerable time may be consumed before a final decision is made to shut down the reactor, if necessary. If the pump trouble as indicated by the alarm is only a symptom of a far more serious incipient failure (such as failure of a pump impeller), an immediate manual scram of the reactor possibly would be the only safe course of action. In this case the separations existing between control of water plant and reactor and between reactors present obstacles which may represent safety hazards.

Some advantage may be gained through installation of a central emergency dispatching station at some central location in the area. This station would have a panel containing all alarms considered necessary to provide for overall safety of the two reactors, and would be manned by at least two dispatchers (more if required). These dispatchers would have the responsibility of determining the seriousness of a fault, finding out what was being done about it, and informing the proper people of the event and/or the proper action which should be taken. This system possibly would speed action in many cases and would also have the following advantages:

1. Protection against consequences of having an operator missing from his post or, for any reason, incapable of performing his duties at a crucial time.
2. Better coordination between KE and KW operation.
3. Added protection against lack of adequate knowledge on the part of new operators as to the system or as to emergency measures.
4. Removal of some routine communication duties from shift supervisors so that they would be more readily available to cope with immediate problems.

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5. Possibly discourage the "operate as long as possible" philosophy when it is not compatible with reactor safety.
6. Removal of the possibility that an operator will inadvertently try to save his equipment to the detriment of reactor safety.

b. Emergency Procedures and Drills

It is considered highly important that the proper action is taken in the case of emergencies; also the effectiveness of many safety actions is dependent upon the speed with which they are taken. In the opinion of the writer, and as is shown by military experience, the only assurance that a person, or group of persons, is capable of performing satisfactorily in an emergency is in knowledge that satisfactory performance has been accomplished many times in simulated emergencies. In fact, the same philosophy applies here as in proving equipment reliability by testing, but to a much greater extent. A careful study should be made of possibilities for actual emergency drills which would be formulated in such a fashion that knowledge of all important emergency procedures would be demonstrated and that the procedures were actually carried out. These drills should be initiated in such a manner that they were totally unsuspected by all critical personnel.

GENERAL NOTES ON RELIABILITY AND SAFETY

Many of the studies suggested in the foregoing paragraphs fall into the realm of basic research. It should be emphasized, however, that operation of the K water systems (and also some of the older systems) is beyond the realm of industrial experience; this applies to size of pumping units and pipe velocities, among other things. Where little or no experience is available, the reliability of a system and its components is difficult, if not impossible, to evaluate unless measures are taken to study in detail the conditions prevailing under set modes of system operation. When matters of safety are involved, changes in modes of operation of a system should not be made until the effects of these changes can be evaluated to the extent that safe operation can be insured.

In this light, it is well to consider the present manner of operation of the water plants from the standpoint of the margins of safety provided during the various modes of operation. According to ordinary engineering standards, in cases where safety is involved, design and operation are conducted according to codes which require generous "factors of safety". Operation with little or no margin of safety is permitted only in cases where all pertinent factors related to safe operation are known. An interesting parallel exists between structural design of the 105 or 165 buildings and the flow available to a reactor during the worst power loss scram conditions. In the first case, the factors of safety are high; in the second case they are less than unity. It would appear that less reliance would be placed upon backup systems if primary systems had capabilities greatly exceeding conservative estimates of the requirements, and that the backup systems themselves should be considered "safe" only when designed on the same basis.

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The principal requirement to insure safe operation is to first establish safe limits of operation. These limits should include generous margins of safety in all cases where lack of knowledge or experience prevails. Safety criteria based upon these limits, and formulated in a manner such as those referred to in this report, should be adopted and rigidly enforced to the extent that all operation would be performed within these limits. It is the writer's understanding that this is the reactor safety goal of the Hanford Operation.

5.0 ACKNOWLEDGEMENT

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BIBLIOGRAPHY

1. HW-39819 Time and Water Available At 100-K Reactors With Loss of River Pumps, R. E. Johnson, November 3, 1955.
2. HW-60659 Report of Preliminary Testing, Production Test IP-234-AE - Project CT-775 - Prototype Testing of the 190-K Process Pumping Units, M. H. Schack, et. al., June 15, 1959.
3. Personal communication from KE Shift Supervisor, C. E. Nelson, on September 13, 1959.
4. HW-46000K Process Standards - Reactor, General Electric Company, Hanford Atomic Products Operation, Irradiation Processing Department, Research and Engineering Operation, August, 1957.
5. HW-61887 Reliability and Safety of the K Reactor Cooling System: Part I Engineering Analysis of the Electrical Power and Control System, F. D. Robbins, September 22, 1959.
6. Drawing No. H-1-25523, C. T. Main, March 7, 1955.
7. Power Operations - Operating Manual, Irradiation Processing Department.
8. HW-27155 Process Standards - Water Plant, Irradiation Processing Department, Rev. 1 Research and Engineering Operation.
9. HW-42102 Miscellaneous Water Flow Data and Process Unit Elevations, W. D. Bainard, March, 20, 1956.
10. HW-39435 Design Analysis Activities Report for September, 1955, G. L. Locke, October 21, 1955.
11. HW-41503 Status Report - K Area Water Program, G. O. Amy, February 27, 1956.
12. HW-42964 Final Report: Production Test 105-14-MR, Measurement of Heat Output of KE Reactor after Shutdown and Process Test, MR-105-30, Determination of Minimum and Transient K Reactor Emergency Coolant Flow, P. C. Walkup and R. E. Hall, May 15, 1956.
13. HW-61523 KE and KW 'Last-Ditch System Capabilities', J. A. Roberson, August 10, 1959.
14. HW-59696 K Water Plant Improvements, R. E. Trumble, et. al., March 19, 1959.
15. HW-28033 Monthly Report of Design Analysis Group, April, 1953, G. M. Roy, May 15, 1953.
16. HW-59190 K Water Plant Reliability Task Force Study Committee 4b: Addition to K Plant Systems, R. E. Trumble, February 9, 1959.

DECLASSIFIED

HW-61886

17. HW-25892 Summary Report of Reactor Hazard for Twin 100-K Area Reactors, October 10, 1952.
18. HW-57287 Coolant Water Transient in K Reactor Following Line Rupture, August 27, 1958, J. A. Roberson and S. S. Jones.
19. HW-26687 100-K Area Water Plant Design, Hanford Works, December 18, 1952.
20. HW-45648 100-K Area Water Plant Expansion Study Feasibility, M. H. Schack, September 26, 1956.
21. HW-61763 Reliability of K Water Plant, R. S. Bell, December 16, 1958.
22. HW-57287 Coolant Water Transient in K Reactor Following Line Rupture, J. A. Roberson and S. S. Jones, August 27, 1958.
23. HW-53471 K Reactor One and Two Pump Tripout Tests, S. S. Jones, November 4, 1957.
24. HW-42374 Emergency Generator Failure at Hanford, L. H. McEwen, March 28, 1956.
25. HW-39237 An Analysis of 100-K Process Pump Plant, W. J. Love, October 30, 1955.
26. Model Studies of Suction Lines for Primary Process Pumps 4 and 5 in Building 190-K of the Hanford Atomic Energy Plant, Charles E. Behlke, and Harold D. Pritchett, Engineering Experiment Station, Oregon State College, Corvallis, Oregon, February, 1959.
27. Centrifugal and Axial Flow Pumps, A. J. Stepanoff, New York, Wiley, 1957, 462 pages.
28. ATP 1405 Acceptance Test Procedure 1405, Rev. 1, Control Buildings, West Area, C. T. Main, Inc., February 1, 1955.
29. Verbal discussion with S. S. Jones, September 10, 1959.
30. HW-61617 Flow Reduction in K Reactor Due to Opening of V-73 Valve, J. A. Roberson, August 17, 1959. (To be published.)
31. HW-41398 100-K Area Backup Cooling System, February 12, 1956.
32. HW-61658 Requirements for K Water Plant Backup System, (CGI-844), R. E. Trumble, August 24, 1959.
33. HW-61045 CGI-844 - 100-K Coolant Backup System Scope Requirements, D. F. Watson, July 28, 1959.
34. Conference with S. S. Jones, September 19, 1959.

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A P P E N D I X A

FIGURES

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FIGURE 1

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EFF 90 80 70 60
BHP 50-1200 40-1000 30-800 20 10 0

FIGURE 2
CHARACTERISTIC CURVES - LOW LIFT PUMP

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100
90
80
70
60
50
40
30
20
10
0

BHP
3000
2000
1000
0

% EFF

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FIGURE 3
CHARACTERISTIC CURVES - HIGH LIFT PUMP

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FIGURE 4
CHARACTERISTIC CURVES
MODIFIED LOW LIFT PUMP
(Production Test IP-234-AE)

FIGURE 5
CHARACTERISTIC CURVES
MODIFIED HIGH LIFT PUMP
(Production Test IP-234-AE)

FIGURE 6
100-K PUMP AND REACTOR CHARACTERISTIC CURVES SHOWING CAVITATION CUTOFF POINTS
FROM CURVES PREPARED BY JL BENSON AND SS JONES

FIGURE 7
SERVICE WATER PUMP CHARACTERISTIC CURVES

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BHP
100
300
200

NW-01885

FIGURE 8
K Reactor Piping Cross-Ties

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FIGURE 9
 NORMAL 6 PUMP OPERATION
 BOTH PLANTS UP

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UNCLASSIFIED

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FIGURE 10
KW DOWN AND PROVIDING
HPC BACKUP FOR KE
WHICH IS UP

100 K WEST WATER PLANT FLOW DIAGRAM
PRINCIPAL RAW WATER, SERVICE WATER AND PROCESS WATER LINES
(EAST PLANT SIMILAR)

FIGURE 11
KW DOWN AND RECEIVING
HPC BACKUP FROM KE

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FIGURE 12
KW DOWN, KE FLUSHING HPCT

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UNCLASSIFIED

DECLASSIFIED

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FIGURE 14
 KW HAS 36 In. CONE VALVE OPEN
 KE OPERATING

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FIGURE 15
K REACTOR FLOW REQUIREMENTS AFTER SCRAM,
AND FLOW BACKUP SUPPLY VS TIME
Inlet Temperature = 20°C Summer
Outlet Temperature = 90°C

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FIGURE 16
TYPICAL K WATER PLANT CONTROL ROOM

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A P P E N D I X B

SUPPLEMENTARY REFERENCES

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CTM-K-102 Report on Reliability of Process Water Pumping Equipment and Power Supply, C. T. Main, Inc., November 15, 1952.

CTM-K-103 Design Computations on Raw Water Lines - 1901-K, C. T. Main, Inc., January 22, 1953.

CTM-K-104 Design Computations on Main Sewer Lines - Process Sewers, Retention Areas and Outfall Structures, C. T. Main, Inc., January 29, 1953.

CTM-K-105 Report on Capability of Process Water Pumping Equipment, R. F. Hall, May 27, 1953.

CTM-K-106 Design Computations on Filter Plants, C. T. Main, Inc., October 27, 1953.

CTM-K-107 Design Computations on Main Pump Houses - 190-K; Control Buildings - 165-K; Oil Tanks - 166-K (Project CA-512-W), C. T. Main, Inc., November 6, 1953.

CTM-K-122RD Design and Construction History of 100-K Area Water Facilities (Project CA-512-W) (HAN-58103), C. T. Main, Inc., February 25, 1955.

HDC 2118 Design Criteria - 100-K Reactor Water Plant - General Description Section 2, C. T. Main, Inc., March 29, 1952.

HDC 2119 Appendix to HDC 2118 - Design Criteria - Section II, C. T. Main, Inc., March, 1952.

HW-21605 Header Pressure Requirements Following An Electrical Power Failure, M. W. Carson, May 9, 1952.

HW-24800 Completion Report - Project CA-512 Vol. II, C. T. Main, Inc.,
103 Vol. 2 June 30, 1957.

HW-24941 Working Committee Minutes No. 5 - Project CG-494 - Clearwell Capacity, E. L. Reed, July 13, 1952.

HW-24988 K Area Minimum Clearwell Capacity, G. L. Locke, July 14, 1952.

HW-25631 Joint Project Committee Meeting - Projects CA-512-W and CA-512-R, J. H. Snyder and E. C. Reed, September 2, 1952.

HW-25791 Design Criteria for Process Piping - Project CG-494, J. J. Griffiths, August 13, 1952.

HW-26151 Project Committee Meeting No. 17 - 100-K Water Plant Design - Project CA-512-W, J. H. Snyder, November 7, 1952.

DECLASSIFIED

HW-61886

HW-26543 Project Committee Meeting No. 20 - 100-K Water Design, Project CA-512-W, J. H. Snyder, December 10, 1952.

HW-26671 An Independent Analysis of the 100-K Water Plant Power and Pumping Stations, G. L. Locke, et. al., December 31, 1952.

HW-27092 190-K Secondary Recirculation Lines, W. L. Pearl, February 10, 1953.

HW-27094 Enriched Operation of K Reactor, W. E. Johnson, February 10, 1953.

HW-27241 Consideration of Additional Water Supply 100-K Area, Project CA-512-W, J. H. Snyder and Andersen, February 27, 1953.

HW-27251 Secondary Pump Modification - 105-K, G. M. Roy, February 26, 1953.

HW-27632 Process Specifications - Reactor Process, R. B. Richards, April 1, 1953.

HW-29055 Design Criteria - 1.2660 - Crosstie Piping System, T. F. Robinson and J. E. Love, August 4, 1953.

HW-29749 100-K Process Pump Performance Using Measured Data - Large Annulus, W. J. Love, November 30, 1953.

HW-31180 Assessment of Disaster Control Systems, D. J. Foley, April 30, 1954.

HW-32021 Maximum Flow Rates Obtainable at the K Areas as Restricted by 190 Pumping and the 105 Downcomer, L. E. Foster, June 3, 1954.

HW-32268 Disaster Safety System for the Hanford Reactors - Auxiliary Water Supply, G. L. Locke, et. al., July 1, 1954.

HW-32568 Static and Dynamic Tests at 100-K Process Water System, W. J. Love, July 27, 1954.

HW-32630 A Bibliography of Reactor Hazards, R. L. Steach, August 2, 1954.

HW-32630 Supl A Bibliography of Reactor Hazards, J. J. Hauth, June 12, 1952.

HW-32777 Initial Operating Conditions for KW Reactor, J. H. Warren, August 17, 1954.

HW-33262 Coolant Supply During Charging Operations at 100-K, R. E. Johnson and S. S. Jones, October 1, 1954.

HW-33269 Basis for Specification on Operation of KW Reactor with Faulty Pump Casings, R. E. Johnson, October 1, 1954.

HW-33368 Water Flow Shutoff in K Pile Tubes After Pile Shutdown, S. S. Jones, October 11, 1954.

HW-33701 Design Analysis Activities Report for October, 1954, R. K. Anderson November 10, 1954.

DECLASSIFIED

HW-61886

HW-33822 100-K Filter Plant Flow Rates, J. M. Skarpelos, December 1, 1954.

HW-33833 Conditions for Setting Panellit Gages, K. W. Hess, November 18, 1954.

HW-33870 Heat Generation and Total Heat Output from the Pile After Shutdown, S. S. Jones, November 23, 1954.

HW-33871 An Analysis of K Pile Water Flow Requirements During Shutdown, S. S. Jones, November 23, 1954.

HW-35701 Process Change Authorization RP 14 - Special Limitations for Initial Operation of KW Reactor, O. C. Schroeder.

HW-36013 Operational Aspects of KW Reactor Activation, D. S. Lewis, April 1, 1955.

HW-36180 Process Change Authorization on RP-21-KE Reactor Startup and KW Reactor Limitations, O. C. Schroeder, April 11, 1955.

HW-36549 A Final Examination of the Feasibility and Safety of the Method for Effecting an Emergency Shutdown of the 1706 KER Loop, F. E. Tippets, June 30, 1955.

HW-36621 Recommendations by Advisory Committee on Reactor Safeguards, O. H. Greager, May 6, 1955.

HW-36821 Design Analysis Activities Report for March and April, 1955, R. K. Anderson, May 15, 1955.

HW-38068 A Review of the Emergency Water Supply Systems, N. V. Starkebaum and G. H. Dyer, July 15, 1955.

HW-38211 Interim Report, Reactor Transient Water Requirements - K Pile, G. L. Locke, July 20, 1955.

HW-38373 A Summary of K Reactor Flow Test Data, P. C. Walkup and K. W. Hess, August 8, 1955.

HW-38944 A Method of Solution of Reactor Transient Water Requirements as Applied to the 100-K Plant, M. B. Larson, September 7, 1955.

HW-38959 KW Reactor Operation Subsequent to Replacement to Front Face Connectors, September 15, 1955.

HW-39044 KW Reactor Operation Subsequent to Replacement of Front Face Connectors, P. H. Reinker, September 16, 1955.

HW-39487RD 100-K Shutdown Flows and Power Levels, C. F. Quackenbush, October 10, 1955.

DECLASSIFIED

HW-61886

HW-39643 K Reactor Power Plant Level Limit Based on Emergency Water Flow, K. W. Hess and R. E. Hall, October 28, 1955.

HW-39657 Aspects of the Failure and Repair of 100-K Areas Sewer Outfall Lines, R. S. Bell, October 21, 1955.

HW-39869 Safety Interlocks on Companion Reactors, W. C. Poe, October 28, 1955.

HW-40413RD Performance of KE and KW Reactors from Startup to Present. A. A Janos, November 29, 1955.

HW-40442 BPA Power Disturbance - November 30, 1955, 100-K Area.

HW-40462 Determination of Minimum and Transient K Reactor Emergency Coolant Flow Rates, Process Test MR-105-30, P. C. Walkup, January 4, 1956.

HW-40473 Additions to Shutdown Process Flow Tests at the KE - KW Reactors Development Test No. 105-619-A. S. S. Jones, December 22, 1955.

HW-40691 Project Proposal: Steam Driven Auxiliaries - Buildings 165-KE and KW, January 17, 1956.

HW-40954 Technical Note on K Water Plant Reliability, L. H. McEwen, February 1, 1956.

HW-41030 Design Analysis Activities Report for November and December, 1955, G. L. Locke.

HW-41381 Hydraulic Behavior of the 100-K Effluent Lines, H. S. Davis, January 27, 1956.

HW-41402 Complete Review and Updating of the 100-K Pumping Plant Performance, D. C. Condotta, February 13, 1956.

HW-41404 Scram Shutdown Coolant Requirements for the 100-K Reactors, Calculations for HW-41398RD, H. R. Kosmata, February 14, 1956.

HW-41427 Design Analysis Activities Report for January, 1956, G. L. Locke, February 15, 1956.

HW-41495 Reactor Incidents at Hanford, April 1, 1956.

HW-41903 "Close-Shave" Incident for Discussion at Joint USA-UK Meeting on Reactor Hazards, Davis and Garcia, March 13, 1956.

HW-42279 Extent of K Crosstie Protection, O. C. Schroeder, March 29, 1956.

HW-42541 Technical Note on Crosstie Limit Calculations, L. C. Bupp, April 6, 1956.

HW-42709 Water Flow Aspects of Proposed K Reactor Power Level Increases, O. C. Schroeder, April 10, 1956.

DECLASSIFIED

HW-61886

HW-42970 Transient Heat Output of a K Reactor - A Correlation of Experimental and Theoretical Data, D. L. Condotta, May 8, 1956.

HW-42985 Interim K Pile Limits, L. H. McEwen, October 7, 1955.

HW-42986 105-KE Power Levels and Flow Requirement Forecast, H. T. Wells, May 2, 1956.

HW-43117 Water Flow Requirements of the K Reactor for the Period 1956 - 1960, D. L. Condotta, May 18, 1956.

HW-43466 Post-Completion Report - Project CA-512, 1952 Hanford Expansion, R. S. Bell and J. P. McCollough, June 1, 1956.

HW-43632 A Proposed 100-K Area Emergency Power and Water Pumping Plan, R. C. Hoffman, June 13, 1956.

HW-43827 Coolant Flow Rates at the K Reactors, T. D. Dow and P. C. Walkup, June 29, 1956.

HW-44708 Hazards Summary Report - Projects CG-558 and CG-600 Reactor Plant
Vol. 1 and 2 Modification, R. E. Trumble, December 21, 1956.

HW-45070 Water Treatment History at Hanford Reactors Through August 20, 1956, August 21, 1956.

HW-45611 Discussion of the Economic Justification for Expansion of the K Reactor's Water Plant.

HW-45684 The Hanford Reactor, E. R. Astley, et. al., September 27, 1956.

HW-48876 K Pile Water Plant Expansion, K. W. Hess and L. W. Lang, March 5, 1957.

HW-49012 Effect of Outlet Water Temperature and Tube Corrosion on Hydraulic Demand Characteristics of Process Tubes, J. F. Calkin, March 18, 1957.

HW-49109 Six-Pump Flow Test Results - 100-KW, J. H. M. Miller and R. E. Hall, March 25, 1957.

HW-49161 Notes on Decisions Involving Reactor Hazards, A. B. Carson, K. W. Hess and L. H. McEwen, January 7, 1957.

HW-49210RD Full Speed 5-Pump Test, J. L. Benson, March 21, 1957.

HW-49651 Reactor Flow Capabilities, K. W. Hess, April 17, 1957.

HW-50183 Summary of K Reactor Outlet Temperature Limits, R. E. Hall, May 13, 1957.

HW-50204 K Reactor Water Plant Expansion, D. L. Condotta, June 7, 1957.

DECLASSIFIED

HW-61886

HW-50656 Results of 190-KE Secondary Pump Cavitation Test, J. L. Benson, June 7, 1957.

HW-51329 Metal Performance and K Pile Water Plant Expansion, L. W. Lang, July 12, 1957.

HW-51750 Water Plant Expansion - 100-KE and KW Project Proposal CG-775, D. E. Newby, September 6, 1957.

HW-51970 K Reactor's Hydraulic Demand, K. W. Hess, August 16, 1957.

HW-52059 Pressure Flow Characteristics of the 190-105K Systems for Use With the 100-K Area Water Plant Expansion Studies, E. R. Rudock, August 20, 1957.

HW- 52793 Reactor Bulk Outlet Temperature Limits, S. S. Jones, March 6, 1958.

HW-53439 K Reactor Plant Expansion Analysis, W. A. Massena, November 6, 1957.

HW-53593 Results of Experimental Tests Simulating Supply Pressure Decrease in a K Process Tube, K. G. Toyoda and J. F. Calkin, November 13, 1957.

HW-54126 Additional Charted Material for Nuclear Safety Review, R. E. Trumble, May 1, 1959.

HW-54329 Results of Transient and Steady-State Experiments Investigating Hazards of Flow Reductions in a K Process Tube, G. M. Hesson and W. L. Thorne, January 2, 1958.

HW-54594 PT-IP-1391 100-K Hi-lift Pump Cavitation Test, J. L. Benson, January 20, 1958.

HW-54644 K Primary Pump Power Cross Feed, K. W. Hess, January 21, 1958.

HW-54887 Nuclear Safety Review - KE and KW Reactor Operation, R. S. Bell, February 12, 1958.

HW-54983 Special Incident Report: Low Flow on Crossheaders - KW Reactor, February 5, 1958, R. S. Bell, February 12, 1958.

HW-55379 Water Shutoff to K Area Tubes, N. E. Fuller and S. S. Jones, March 18, 1958.

HW-55422 100-KW Raw Water System Characteristic Curves, N. F. Fifer, March 21, 1959.

HW-55877 Recommendations for 100-K Area CG-775 Raw Water Requirements, N. F. Fifer, April 29, 1958.

HW-56131 Sulfuric Acid Piping Failure at 183-K. W. D. Bainard, May 15, 1958.

DECLASSIFIED

HW-61886

HW-56269 System Curves for 100-K Water Plant Expansion Pump Analysis, E. R. Rudock, June 5, 1958.

HW-56672 Results of PT-IP-139-1 - 100-K Hi-Lift Pump Cavitation Test, J. L. Benson, July 15, 1958.

HW-56764 Meeting with Advisory Committee on Reactor Safeguards - Hanford June 5 and 6, 1958, and Summary of May 9, 1958 Meeting, R. L. Dickeman, July 14, 1958.

HW-56884 Selection of 181-K River Pump Impellers, M. H. Schack, June 22, 1958.

HW-58499 Performance Curves for 181-K River Pumps with New 31 7/32" Diameter Impellers, N. F. Fifer, December 11, 1958.

HW-58614 K Reactor Water Plant Analysis, R. L. Dickeman, December 22, 1958.

HW-58899 CA 704 100-K Outfall Study: Final Report, D. F. Watson, January 31, 1959.

HW-59115 Total Control VS. Reactivity Parameters, D. E. Simpson, February 5, 1959.

HW-59672 Production Test IP-324-AE Project CG-775 Prototype Testing of the 190-K Process Pumping Units, M. H. Schack and S. S. Jones, March 17, 1959.

HW-59741 100-K Area Coolant Backup, D. E. Newby, March 26, 1959.

HW-61658 Requirements for K Water Plant Backup System, (CGI-844), R. E. Trumble, August 24, 1959.

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A P P E N D I X D

INVESTIGATIONS REQUIRED FOR COMPREHENSIVE STUDY
OF WATER PLANT SAFETY AND RELIABILITY

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INVESTIGATIONS REQUIRED FOR COMPREHENSIVE STUDY
OF WATER PLANT SAFETY AND RELIABILITY

The specific objectives formed by this writer in analysis of the safety and reliability of the water plant were many and varied. In a complete study of such a problem, it was considered that the following investigations were required:

1. Thorough study of the physical arrangement of the system by examination of as-built drawings, physical inspection of the system and conference with operating personnel and engineers closely associated with the system.
2. Careful analysis of operating data to reveal existence of safety hazards or indications of unreliability of equipment.
3. Review of possible operating conditions for which no data or experience exist, and check by independent computations for system adequacy against design computations, if available.
4. Study of individual pieces of equipment or other system components for clues to present or future lack of reliability or to inadequacy; study of equipment operating characteristics.
5. Thorough analysis of the consequences of failure of equipment or other system components in terms of effects both on the cooling system and the reactor.
6. Review of all previous writings dealing with study of the system and its reliability and safety.
7. Examination of operation procedures, operating standards for defects such as inconsistencies, omissions, errors or philosophies which could have hazardous consequences.
8. Study of alarms, safety circuits, automatic trips, emergency procedures, communications facilities and procedures and overall control of the reactor and water system for omissions, inadequacies or hazardous conditions; study of personnel training methods for deficiencies and interview of operating personnel for detection of the existence of unsafe attitudes or conflicts or interest amongst power, production and/or management.
9. Evaluation of results of the foregoing listed studies in terms of safety criteria in order to determine the necessity for remedial action required in system modifications or in operating restrictions.

A complete study along the lines of the foregoing objectives was, of course, impossible to accomplish in the time available; furthermore, many aspects of these objectives involved highly specialized fields of study. Consequently, this writer selected from the list those items considered most appropriate for the time available and most commensurate with his technical background.

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A P P E N D I X E

SAFETY CRITERIA - REACTOR COOLING SYSTEM RELIABILITY

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SAFETY CRITERIA - REACTOR COOLING SYSTEM RELIABILITY

"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. Therefore, the extent of the unusual measures which will be required in a reactor plant to achieve an acceptable level of cooling system reliability will be a matter of judgment; the 'ultimate reliability' of a system or any series of systems cannot be expressed quantitatively nor can an 'acceptable level of cooling system reliability' be quantitatively defined - 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. However, even though any system or combination of systems can fail, it is necessary 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.

"The normal reactor cooling system requirements shall be:

"1. Three independent, reliable sources of cooling as follows:

"a. 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.

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"b. 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 cooling equipment. That flow provided by the secondary system during normal operation may contribute to the primary cooling system requirement.

"c. Last-Ditch System

The last-ditch cooling system must be independent of both the primary and secondary cooling systems, including piping to the reactor manifolding. 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.

- "2. Any 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.
- "3. 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.
- "4. 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.
- "5. All engineering, construction, operation and maintenance shall meet or exceed requirements specified by applicable codes and standards."

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