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February 4, 1960

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HAZARDS SURVEY
1706-KER RECIRCULATING TEST FACILITY

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February 4, 1960

HAZARDS SURVEY
1706-KER RECIRCULATING TEST FACILITY

by

K. W. NORWOOD

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ABSTRACT

This report presents the results, conclusions and recommendations of a comprehensive study of the hazards potential of the 1706-KER in-pile recirculation loops. Although more data are needed to verify or refute certain tentative conclusions, the general conclusion of this report is: with the exception of severe coolant loop rupture incidents, especially process tube ruptures, the loops will not possess serious hazards potential when they are operated within the limits specified in the Process Standards as amended by the acceptance of the recommendations of this report concerning the pressurizer vent system and the process tube exit trips.

The potential for fuel burnout or meltdown following a coolant loop rupture should be reduced considerably after completion of Project CGI-839, "Modification to Fuel Element Test Facilities".⁽¹⁾ The present potentially "worst case" coolant loop rupture could lead to an accident comparable in severity to a single K-Reactor tube rupture and meltdown. Serious personnel and property contamination would be confined to the vicinity of the reactor following completion of Project CGI-791, "Reactor Confinement".⁽²⁾ unless unusual atmospheric conditions persisted during the entire release time, i.e., strong, uni-directional winds causing rapid straight-line, non-mixing travel of the fission products plume. However, graphite stack damage from a tube rupture could be severe and costly.⁽³⁾

Coolant system reliability criteria for non-production facilities located in Hanford production reactors are now being prepared. In the absence of such criteria, the Reactor Coolant Supply System Reliability Criteria were applied to the KER loops in this report. However, it should be noted that, if the proposed non-production facilities criteria are accepted in their present form, the basic statements concerning the adequacy of the KER loops' coolant systems given in this report will not be altered significantly.

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INTRODUCTION

The 1706-KER Loops consist of four (4) high-temperature, high-pressure, through-pile loops located at KE Reactor. The loops were built for the purpose of testing a large variety of possible future reactor fuel elements under prototype conditions of temperature, pressure, coolant flow rate, water quality and reactor flux. The basic physical layout of a primary loop and a line diagram of a #1 safety circuit are shown in Figure 1. Present loop operating limits are given in Table I. Detailed descriptions of the physical features and instrumentation of the loops are given in references (5) and (6).

A detailed hazards study of the KER loops was deemed necessary for two reasons: 1) Safe operation of the loops is essential to the welfare of the loops themselves, the KE Reactor and their respective operating staffs and 2) The many similarities between a KER loop and a more complex NPR loop indicate that a better understanding of the hazards potential of a KER loop should be of significant aid in designing safeguards for the NPR.

SUMMARY

A number of possible accidents with potentially serious consequences were studied:

1. Fuel element burnout or melting,
2. Opening of a dump valve during recirculation operation,
3. Loss of coolant flow to the loop primary heat exchangers,
4. Coolant loop rupture, including process tube rupture,
5. Loss of one or more recirculation pumps,
6. Rupture of a pressurizer rupture disk, and
7. Reactor power excursion.

In addition, the subjects of potential fission products release, the adequacy and reliability of a loop safety circuit and the compliance of the loops with the "Safety Criteria - Reactor Coolant System Reliability" (4) were investigated.

This report presents a discussion of these safety aspects of the 1706-KER loops and the conclusions and recommendations which were the ultimate result of the study.

CONCLUSIONS

The conclusions resulting from this study can be summed up in the following statement:

With the exception of severe coolant loop rupture incidents, especially process tube ruptures, the loops will not possess serious hazards potential when they are operated within the limits specified in the Process Standards as amended by the acceptance of the following recommendations concerning the pressurizer vent systems and the process tubes exit trips:

1. The following changes in the pressurizer vent piping and operating procedure should be made in order to minimize the hazards potential

MAIN SCHEMATIC DIAGRAM OF LOOP PROCESS AND SAFETY
CIRCUIT SYSTEMS

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TABLE I

PRESENT 1706-KER LOOPS OPERATING LIMITS

General

Zirconium Tube Temperature, High, °C	350
Cold Hydrostatic Test Pressure, New Tube, psig	4400
" " " " , Subsequent Tests, psig	4000

Single Pass Operation

Process Tube Outlet Temperature, High Trip, °C	100
--	-----

Recirculation Operation

Process Tube Inlet Flow, High Trip, gpm	68
Process Tube Outlet Temperature, Operational, High, °C	

Loop 1	275
Loops 2, 3, 4	293

Process Tube Outlet Temperature, High Trip, °C

Loop 1	285
Loops 2, 3, 4	303

System Pressure, High, psig

Loop 1	1350
Loops 2, 3, 4	1600

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resulting from bursting of a rupture disk or other severe pressurizer steam leak and to bring the venting system into compliance with the requirements set forth in Section VIII, UG126-d-4 and Ruling 1-a, Case 1271N, of the ASME Boiler and Pressure Vessel Code:

- a) The procedure of locking open the vent valve in the free vent line between each rupture disc-relief valve combination should be replaced by a procedure of locking the valve closed.
- b) A pressure gage with a suitable high pressure alarm should be installed in the section of vent piping between each rupture disc-safety relief valve combination. This gage must be capable of withstanding full system pressure during an emergency.
- c) An additional rupture disk-relief valve combination in a separate vent line should be installed on each "side" of the vent system "downstream" from the 3-way valve.
- d) The following additions to the Process Standards are recommended:
 - i) The valve in the vent line between each rupture disc-safety relief valve combination must be locked in the closed position during recirculation operation.
 - ii) The high pressure alarm on the gage which monitors the pressure in the piping between each rupture disc-safety relief valve combination shall be activated by pressures equal to or greater than 15 psig. If the high pressure alarm is activated during operation, the gage reading must be monitored at 15 minute intervals. If the gage reading does not exceed 25 psig, recirculation operation may be continued until the next reactor shutdown, at which time the pressureizer pressure must be reduced and the offending rupture disc must be inspected for leaks.

If the gage reading exceeds 25 psig at any time for a period of 5 minutes or more, the loop pressure and tube exit temperature must be reduced to 500 psia and 130°C, respectively, until the next reactor shutdown, at which time the offending rupture disc must be inspected for leaks.

In all excessive pressure cases, continued operation of the affected loop on recirculation operation following disc inspection is not permitted unless the offending rupture disc is first replaced or a "Process Change Authorization" is obtained from Process Technology Sub-Section.

- iii) The system pressure (tube exit) and boiling point suppression low trips shall be set so that their contacts will open at or above that one of the following pressure which is the highest:

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Chosen equilibrium operating pressure minus 150 psi, or minimum pressure specified in production test, or pressure resulting from specifications of Process Standard K-040.02 B&C.

In addition, the safety relief valves should be tested and adjusted to prevent excessive pressure reduction below their respective settings from failure to reseal properly and rapidly.

2. The process tube exit trip circuits (high temperature, low pressure, low boiling point suppression) should be altered so that:
 - a) The failure of a single component will not scram the reactor in the absence of a scram condition,
 - b) The failure of a single component will not prevent a scram in the presence of a scram condition.

The following conclusions represent the bases for the general statement given above and for the recommendations which follow certain of the conclusions:

1. If present practices concerning the determination of fuel element test limits are followed, burnout of the fuel element(s) will not occur under equilibrium or transient coolant flow conditions with the following exceptions:
 - a) It is possible that limited fuel burnout might occur during the pressure, flow and heat flux decays following opening of a dump valve during high-pressure, high-temperature recirculation operation. Although the small amount of available data indicates that burnout should not occur, a definite statement on this problem cannot be made until more data become available.
 - b) Fuel burnout will occur if a coolant loop is ruptured severely. A "worst case" loop rupture could lead to an accident which would be comparable in fission product release potential to a single K-Reactor tube meltdown, with additional potential damage to the graphite stack.
2. With the possible exception of the case where both recirculation pumps in a loop are lost simultaneously, credible pump loss incidents will not lead to reactor or loop hazards. The excepted case represents a variation of the dump valve opening incident, and as in conclusion 1-a, while the limited data indicates that fuel burnout will not occur a firm conclusion must await more data.
3. No damage to the loop, the fuel elements, or the reactor will result from the loss of coolant flow to the secondary side of the primary loop heat exchangers if the heat exchanger exit temperature high trip setting is set as specified in HW-46000 K, "Process Standards-Reactor".

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4. With the present loop instrumentation and procedures of operation, it would be possible within the provisions of the pertinent Process Standards for inexperienced personnel to set a loop's system pressure and boiling point suppression low trips such that the bursting of a pressurizer rupture disc or a severe steam leak from the pressurizer auxiliary equipment could lead to fuel element burnout.
5. The loop pressure sensing system (tube exit pressure and pressurizer pressure) do not conform to the recently amended B&PV Code in that the two independent sensing devices are not powered by independent external energy sources.

Recommendation: The pressurizer pressure sensing and recording system should be powered by electrical energy from 165-KE while the tube exit pressure and boiling point suppression systems should be powered by energy from 165-KW or vice versa.

6. An extremely small probability exists that the combination of poorly chosen operating limits and the accidental, simultaneous withdrawal of two horizontal control rods could lead to fuel burnout. Adoption of a proposed method of controlling reactor power surges⁽⁸⁾ should reduce this probability essentially to zero.
7. Following completion of Project CGI-791, "Reactor Confinement",⁽²⁾ property damage caused by the fission products release from a single KER tube meltdown should be confined to the vicinity of the reactors. Release of fission products to the river will occur only if the reactor effluent stream is not diverted to a suitable crib for storage.
8. Personnel should continue to exercise extreme caution during retubing, fuel charge-discharge, and fuel element rupture discharge operations to prevent damage to the process tubes (scratching, gouging, thread damage, etc.) in an effort to minimize the potential for process tube rupture.
9. The pressurizer pressure relief system does not conform with the recently amended ASME Boiler and Pressure Vessel Code (hereafter abbreviated as B&PV Code) in that at least two relief valves are required to be in operation at all times.⁽⁷⁾ (See page 40, item 4.)
10. The #1 Safety Circuit instrumentation of the loops is reliable and appears adequate with the exception of the process tube exit trip circuits. (See Figure 1.) These trips do not have a reliable back-up elsewhere in the safety system, so they cannot be operated using parallel circuits.

Recommendation: A two out of three trip matrix approach should be employed. In addition, the following changes in the 1706-KER Safety Circuit also are recommended for detailed study and evaluation with a view toward eventual adoption:

- a) Installation of a two out of three matrix with three independent circuits and temperature sensing devices in place of the single circuit and sensing device now used to measure the heat exchanger exit water temperature and to provide a high trip.
 - b) Connection of the recirculation pump differential pressure trips and the loop low flow trips for each loop in a matrix such that:
 - i) One Δp trip plus one low flow trip will cause a reactor scram, but
 - ii) Two Δp trips will not cause a reactor scram.
 - c) As a less desirable alternate to change b, revision of the Process Standard K-040.02-H so that the required Δp trip setting is about 60% of equilibrium flow Δp at cold flow conditions ($\sim 65^\circ\text{F}$) instead of 67% as at present.
 - d) Installation of an electrical intertie incorporating an alarm between the dump valve selector switch and its by-pass switch (By-Pass Switch #111) so the dump valve selector switch cannot be turned from "Automatic" to "Recirculate" unless it is first by-passed. The alarm shall be actuated when the circuit is un-by-passed.
11. The KER loops do not comply with the "Criteria - Reactor Coolant System Reliability"⁽⁴⁾ in the following respects:
- a) The primary coolant system is inadequate to allow safe transition to the secondary and last-ditch systems in the event of a severe coolant loop rupture, and possibly in the cases of accidental dump valve opening and simultaneous loss of both pumps in a loop.
 - b) Failure of the primary system through loop rupture also can lead to failure of the secondary and last-ditch systems.

Planned revisions to the loops⁽¹⁾ should remove most of the problems. However, additional unplanned revisions such as provision for additional independent pump power supplies may be required to accomplish complete compliance with the Criteria.

DISCUSSION

I. Description of Facilities and Justification for Study

The KER Loops are high-pressure, high-temperature recirculation facilities located at KE-Reactor. They were built for the purpose of testing a large variety of possible future reactor fuel elements under controlled conditions of temperature, pressure, coolant flow rate, water quality, and reactor flux. A schematic diagram of a main loop and its #1 Safety Circuit is given in Figure 1. Table I contains a list of loop operating limits. There are four ⁽⁴⁾ through-pipe, recirculation loops in the overall facilities. No. 1

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Loop is constructed of Schedule 160 carbon steel piping, while Loops No 2, 3 and 4 are constructed of Type 304 stainless steel piping. All four loops contain Zircaloy-2 process tubes. The four loops are essentially identical except for the aforementioned difference in material of construction, which results in the following approximate total piping volumes for the loops (exclusive of pressurizer):

<u>Loop</u>	<u>Volume of Loop, Gallons*</u>
1	154
2	220
3	226
4	230

*An average value of 225 gallons was used in all calculations involving Loops No. 2, 3, and 4.

The physical layout of a KER Loop is described in detail in Reference 5. The functions and details of the instrumentation are given in Reference 6.

There were two primary reasons for making this hazards study:

- a) These test facilities are an integral part of the KE-Reactor complex. Therefore, their safe operation is essential to the welfare of the loops themselves, their operating personnel, and KE-Reactor.
- b) A KER Loop has many of the features of an NPR Loop. Therefore, a better understanding of an evaluation of the hazards involved in KER loop operation should aid in planning safeguards for the NPR.

II. Descriptions and Analyses of Possible Incidents

A. General Bases for Analyses and List of Potentially Serious Incidents

The first step in analyzing a system for hazards potential is to study the system and assemble a list of all possible incidents which could occur and which might lead to serious consequences. One must then reduce this list to those incidents which are considered to have a finite and reasonable probability of occurring during the lifetime of the system. Admittedly, serious errors of judgment can be made in this last step unless care is exercised, especially if the consequences of the incident are severe, as in a nuclear system. Therefore, one must make every effort to analyze each incident situation from a reasonable but always conservative viewpoint.

In analyzing the KER Loops, the following assumptions were made:

- a) Occurrence of combinations of two or more simultaneous, unrelated loop component failures were not considered because of their extreme low probabilities of occurrence. However, simultaneous failures of a loop component and a safety trip were considered.

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- b) Incidents involving gross KE-Reactor damage as well as damage to the KER facilities, such as a complete loss of all coolant to KE-Reactor during single-pass KER operation or severe graphite stack damage resulting in tube ruptures and large-scale emission of fission product to the atmosphere from both KE and KER tubes, were not considered in this study since contamination from and damage to KE-Reactor would be far more serious than contamination from and damage to the KER facilities. Therefore, protecting the loops against these incidents would not result in any significant improvement in nuclear safety. Essentially, the second assumption eliminates severe earthquakes, tornadoes, floods, sabotage, and bombing from consideration.

The incidents which are treated in the following discussion are:

- a) Fuel element burnout,
- b) Inadvertent opening of dump valve, reactor operating,
- c) Loss of cooling water to heat exchanger,
- d) Coolant loop rupture, including process tube rupture,
- e) Pump losses,
- f) Bursting of pressurizer rupture disk, and
- g) Power surges from inadvertent poison column discharge or rod withdrawal.

B. Thermal Hydraulic Incidents

1. Fuel Burnout

1.1 General

The phenomenon of fuel element burnout or melting is of considerable interest and importance in the nuclear field. The subject is rather poorly understood theoretically. However, numerous empirical correlations exist in the literature which can be used to predict burnout heat fluxes for a variety of operating conditions^(10,11,12). With the use of appropriate degrees of conservatism in applying these correlations to actual fuel element cases, it is possible to stipulate operating conditions for any fuel charge which will assure safe equilibrium operation from the standpoint of burnout consideration.

In a high temperature, high pressure recirculation facility such as a KER Loop, the strength of construction of the facility imposes a maximum allowable working pressure limit on the operation of the loop. In addition, the desire to prevent bulk boiling because of the uncertainty in burnout calculations automatically imposes a maximum temperature limit. However, the design of the fuel elements and the tube powers and heat fluxes involved in most tests are such that operation of the loop at these maximum conditions would lead to fuel burnout. Therefore, in practice, a given fuel charge

is usually tested within operating limits of maximum allowable temperature and minimum allowable flow rate and system pressure which are dictated by burnout limits.

The following sub-sections will treat burnout considerations for the four possible modes of operation of a KER Loop: equilibrium recirculation, equilibrium single-pass, transient "recirculation", and transient single-pass.

Of all the elements tested in the loops, the seven-rod cluster element has the highest ratio of critical* channel ΔT to tube bulk ΔT . Therefore, all "worst case" calculations concerning burnout in this report were based on the seven-rod cluster geometry.

1.2 Equilibrium Recirculation Operation

Most fuel element tests are designed for recirculation operation. For these cases, the operating limits are determined by two criteria, both of which must be satisfied:

- a) There shall be no burnout with the loop operating at the maximum allowable tube outlet temperature (high temperature trip), the minimum allowable tube flow rate (low flow trip) and the minimum allowable system pressure.
- b) The temperature of the coolant at the critical flux point* in the tube shall be equal to or less than the saturation temperature at that point for equilibrium operation. Local boiling is permitted at the maximum tube outlet temperature, minimum tube flow and system pressure conditions provided burnout conditions are not encountered.

A sample recirculation operation burnout calculation is given in Appendix C. The results of this calculation are summarized in Table II.

1.3 Equilibrium Single-Pass Operation

Very few fuel element tests are designed for single-pass operation. However, reactor shutdown and subsequent loss of production from equipment failures during recirculation operation can be prevented in some cases by switching the affected loop to single-pass operation. Therefore, in planning a recirculation test, one should determine acceptable single-pass operating conditions as a safety precaution. A "worst case" sample burnout calculation for single-

*The critical flux point is that point in the fuel charge where burnout conditions would be encountered first as one increased the bulk coolant temperature at constant tube power and flow rate. For a multi-channel element such as a tube-and-tube or seven-rod cluster element, the flow channel in which this critical flux point would be encountered is called the critical flow channel. The term "critical" as used in this discussion of burnout does not imply critical flow as normally understood.

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pass operation is given in Appendix C. Inspection of the results of this example, which are summarized in Table II, reveals that the margin of safety in equilibrium single-pass operation would be very large in all cases, unless a new fuel design with a considerably higher critical channel to bulk tube ΔT ratio were conceived. However, it is doubtful that a design radical enough to lead to burnout problems during equilibrium single-pass operation would be of any practical use even in recirculation systems.

1.4 Transient "Recirculation" Operation

The transient "recirculation" cases of interest are:*

- a. Radiation trip causing reactor shutdown. Flow remains constant.
- b. Hydraulic incident-caused transients, such as loss of heat exchanger cooling water and loop line rupture. Flow may vary.

One may note that the word recirculation is in quotes in the title of this sub-section since several of the incidents in item b actually cause the mode of operation to change from equilibrium recirculation to a form of single-pass operation.

When a radiation trip is received, the reactor is shut down, and an automatic loop depressurization cycle is initiated. This depressurization cycle is automatically controlled by pre-set valve positions and heat exchanger control action such that boiling and/or burnout will not occur during the shutdown transient.⁽¹³⁾ The radiation trip sequence is adequately treated in Reference 13, so no further discussion is necessary in this report.

The incident-caused transients in item b are treated as separate cases in later sections of this report. In addition, transients caused by reactor power surges for all types of loop operation are discussed under Section II-C.

1.5 Transient Single-Pass Operation*

The "worst case" credible thermal hydraulic incident for the KE-Reactor involves the instantaneous loss of one pump during five-pump operation. Such a loss would lead to a flow reduction to a single-pass KER tube of about 20%. The effects of such an incident on the normal single-pass operation of the "worst case" fuel element are illustrated in Table II. The detailed calculations are given in Appendix C. One can see

* Nuclear transients are considered in Section II-C.

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TABLE II

SUMMARY OF SAMPLE BURNOUT CALCULATIONS

- NOTES: 1. Symbols are defined in Appendix A, page .
2. Operating conditions referred to in this Table are listed in Appendix C, page .

<u>TYPE OF OPERATION</u>	<u>ϕ_R</u>	<u>T_L</u>	<u>$T_{L,S}$</u>	<u>T_O</u>	<u>$T_{O,S}$</u>
Equilibrium Recirculation:					
a. Normal Operating Conditions	1.62	250*	313.	200	313.
b. Tube Exit High-Trip Temperature, Low-Trip Flow Conditions	1.012	265.5	313.	210	313.
Equilibrium Single-Pass	2.205	135.	211.	91.5*	100*
Transient Single-Pass Following Instantaneous Loss of Single Pump	1.022	167.	191.	113*	100*
Transient Recirculation Following Maximum Single-Tube Power Surge					
a. Surge from Normal Conditions	0.863	284.5	313.	22.6	313
b. Surge from Tube Exit High Trip Temperature, Low Trip Flow Conditions	0.947	275.8	313.	216.8	313
Transient Single-Pass Following Maximum Single Tube Power Surge	0.79	158.3	211.	107.2*	100*

*For Single-Pass Cases only, T_O and $T_{O,S}$ refer to outlet actual and saturation temperatures downstream of the dump valve, i.e., in the effluent piping.

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from the results that a large burnout safety factor exists at the worst transient condition. However, the tube exit temperature is above the boiling point for the effluent system pressure. Since it is desirable, though not absolutely necessary, to prevent boiling in the effluent system at all times, determination of the equilibrium flow rate should include consideration of this factor.

Unexplained increases in tube power at constant reactor power of 10 to 30% have been observed to occur when tubes are switched from high temperature recirculation operation to low temperature recirculation or single-pass operation.* This fact should also be kept in mind in determining test limits. Efforts are being made to develop an explanation for the tube power fluctuations.

1.6 Summary

In summary, proper determination of the operating limits for each fuel element production test will insure:

- a) Safe operation under all equilibrium operating conditions, and
- b) No complications during transient operation following a radiation trip during recirculation operation or a single-pump loss incident during single-pass operation.

2. Dump Valve Opening During Recirculation Operation

Two theoretical analyses of the flow transient which would occur upon the opening of a dump valve to the effluent system and atmospheric pressure during high-pressure, high-temperature recirculation operation have been published.^(13,14)

Both of these analyses predict that there will be a flow surge to a peak rate of from 2-3 times the equilibrium rate within 1-2 seconds after the valve opens. This peak will be followed by a slow, steady decrease to a flow rate below normal before the process backup water begins to flow into the process tube. Both analyses lead to the conclusion that the loops can experience a dump valve incident without sustaining fuel element burnout.

Unfortunately, both of these analyses are based on assumptions which have never been tested experimentally. In addition, the following two of these assumptions do not apply to the loops as they are presently operated:

- a. The loop flow control valve will remain in its pre-incident position during the transient.
- b. The pumps will neither aid nor hinder the flow during the transient.

*Personal communication, W. K. Krätzer, Reactor Fuels Unit, R&E, IPD.

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The first assumption does not apply in that the flow control valve will close off to its minimum opening during the flow surge which follows the dump valve opening. Such action will greatly decrease the peak flow rate. In addition, since the transient flow process can be described by an equation of the form:

$$P_{pr} - 14.7 = \Delta P_f + \sum \frac{L_p}{S_p} \frac{dQ}{dt} \quad (1)$$

where the second term on the right side of Equation 1 represents the sum of acceleration and/or deceleration terms, it is evident that the transient flow curve will be expanded or spread out with respect to time. The curves in Figure 2 illustrate these effects. The net result of the control valve action is to cause the initial portions of the predicted curves to be non-conservative while the latter portion will probably be too conservative.

The second assumption is unrealistic in that the pumps will continue to run at full speed during the transient unless an operator happens to turn them off manually. Therefore, under present conditions, the pump head should be included in the overall flow driving force until the net positive suction head to the pumps decreases to the point where pump cavitation begins. Once cavitation begins, it is quite possible that the pumps will severely restrict the flow if they are allowed to run. The only known, useful data on this process confirm these conclusions. (15)(25)

In view of the discrepancies between the analyses and fact and the need for accurate data on this system a production test is being prepared to authorize the conductance of a rapid depressurization test from high temperature, high pressure conditions. The results of this test should be extremely valuable in:

- a. Determining the needed changes, if any, in the present loop design to insure safe shutdown.
- b. Revising analytical procedures to better represent the actual system so that proposed loop revisions⁽¹⁾ can be properly designed to handle this shutdown transient.
- c. Checking out a simplified version of the yet-to-be-completed NPR line rupture computer program to help determine the applicability and confidence limits for the theoretical equations in the main program.

The results of the production test and definite conclusions regarding items a and b above will be published as soon as possible after completion of the test.

In the interim, the following tentative conclusions are offered, based on information in references 13, 14 and 15:

EFFECT OF CONTROL VALVE ON LOOP FLOW DECAY FOLLOWING DUMP VALVE OPENING

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- a. The loops can adequately handle rapid depressurization shutdown transients provided that the recirculation pumps are tripped out of service about 10-20 seconds after the dump valve is opened.
- b. A process tube in good condition appears capable of successfully withstanding a limited number of the thermal shocks associated with the shutdown transient. This topic is more fully covered in reference 15 and will not be treated in this document.

The first conclusion cannot be extended in scope to cover the case of non-trip of pumps due to lack of knowledge concerning cavitation. However, continued operation of the pumps beyond the initial stages of the transient would be detrimental to the pumps themselves and, possibly, to the fuel elements. Therefore, it is recommended that the loop operators be instructed to proceed as follows in the event a dump valve incident does occur:*

- a. Make sure the dump valve remains open.
- b. Turn the pumps off as soon as signs of cavitation, such as widely fluctuating pump differential pressures and motor amperages, are noted.
- c. Leave all other valve settings unchanged from their equilibrium positions until the entrance of process backup water into the loop has been verified by the various pertinent recorders (tube exit temperature, boiling point suppression).

If this procedure is followed, the pumps will aid the flow surge but they should be off the line before they can cavitate seriously and hinder flow.

3. Loss of Secondary "Loop" Coolant to Primary Loop Heat Exchangers

The coolant flow to the primary heat exchangers is supplied to all four loops by a single raw water pump. If this pump fails, area service water will automatically flow into the coolant supply lines through an air-operated valve system. Both the primary pump and the service water pumps are powered by energy from the 4160 volt system in the KE Area.

There are four ways in which coolant flow to the primary heat exchangers could be reduced below normal:

- a. Plugging of supply piping.
- b. Rupture of supply piping.

* This procedure has already been recommended in reference 25. Therefore, it was not included in the "Recommendations" section of this report.

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- c. Loss of 4160 power (BPA and emergency) at KE area.
- d. Malfunctioning of the coolant flow controller and/or flow control valve.

Line plugging is quite unlikely since the smallest line in the supply system is four inches in diameter. Line rupture is also an unlikely occurrence, because line pressures are quite low (~70 psig). However, loss of 4160 power has an estimated probability of occurrence of one in 20 years⁽⁴⁾ and instrument malfunction is an ever-present possibility. In connection with the valve and/or controller mal-function case, one may note that a stem in the valve prevents complete water loss.

In analyzing the subject incident, the following simplifying assumptions are made:

- a. Loss of coolant flow is instantaneous and complete.
- b. The primary loop water in the heat exchanger exhibits a linear temperature gradient from inlet to outlet.
- c. The heat capacity of the system piping and fluid travel time in the process tube are ignored.
- d. The reactor is effectively scrammed 3 seconds after reception of a trip signal.
- e. The equilibrium tube exit temperature prior to water loss equals the maximum allowable value.

These assumptions all lead to calculation of higher coolant temperature peaks than would occur in practice for major reductions in heat exchanger coolant supply. Thus, the use of all five assumptions yields a very conservative case. In addition, their use also yields a much-simplified case for analysis.

At the moment of cooling water loss, there is a long column of already-cooled water between the heat exchanger outlet and the process tube inlet and a long column of uncooled water between the process tube exit and heat exchanger inlet. Spampinato⁽¹³⁾ has estimated the fluid travel time between various locations in the loop at a flow rate of 60 gpm. In addition, it is estimated conservatively that the loop fluid loses enough heat to the surroundings at elevated temperatures (450-600°F) to lower its temperature by at least 20°F during travel from tube exit to tube inlet. This estimate is based on both experimental and analytical information.

The above information and assumptions can be combined with the conservative curve of heat generation of the charge versus time

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after effective reactor scram given in reference 16 to yield a curve of process tube exit temperature versus time after coolant flow loss. Such curves for the following cases in Loops 2, 3, and 4 are shown in Figure 3:

- a. Reactor scrambled by heat exchanger exit temperature high trip, present setting ($T_{e,HT} + \Delta T_{HT} - \Delta T_L$).
 1. 1200 kw tube power, 60 gpm.
 - ii. 200 kw tube power, 60 gpm.
- b. Reactor scrambled by process tube exit temperature high trip-boiling point suppression low trip combination - 1200 kw tube power, 60 gpm.

Calculations were also made for Loop #1. They show that the only differences between the loops with regard to the subject curves are those of different equilibrium temperatures and different elapsed times between cooling water loss and reduction of loop temperatures to safe levels, i.e., to levels at which the dump valves can be safely opened without flashing or fuel burnout in the loops. Therefore, the following observations concerning the curves in Figure 3 are applicable to all four loops:

- a. The oscillations and maximums in process tube exit temperature become more severe as tube power is increased.
- b. The maximum process tube exit temperature after cooling water loss for present loop operation is 579°F (1° over trip) if the accident occurs when the high trip tube exit temperature prevails, just prior to the accident.
- c. Bulk boiling will occur in the process tube if the process tube exit temperature high trip-boiling point suppression low trip combination is relied on for the reactor scram signal.
- d. The affected loops will eventually cool to a temperature at which process back-up flow can be initiated without sustaining flashing or fuel burnout in the loop.
- e. In cases of loss of 4160 volt power, the pressurizer heaters will be inoperable and the pressurizer will slowly cool and depressurize.

One may note that the effect of the heat capacity of the loop piping would tend to severely damp the large temperature oscillations. This damping would lead to a shortening of the time needed for the loop to cool below process backup water saturation temperature. Since conditions will be different depending on whether cooling water is lost due to line plugging, line rupture or valve malfunction, or due to electrical power loss, the remaining discussion of this subject will be divided into two parts.

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FIGURE 3

PROCESS TUBE EXIT TEMPERATURE Vs. TIME AFTER HEAT EXCHANGER
COOLANT LOSS, LOOPS 2, 3, 4.

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First, consider the case of line rupture, line plugging, or valve malfunctioning. One would have a situation where the loop pressure would be maintained at its equilibrium level. Therefore, no immediate corrective action to restore cooling water would be required. Process backup water could be safely admitted to the loop about 40-50 minutes after cooling water loss.

If cooling water were lost through electric power failure, then the pressurizer heaters and the KE-Reactor pumps would also be lost. It is quite possible that this accident might occur while KW-Reactor is shut down. Under these conditions, the loops* must be cooled to 290°F or less before process backup water can be admitted. From the curves in Figure 3, plus a conservative estimate of the decrease in heat losses to the surroundings per unit time as the loops cool, it appears that the loops would cool to 290°F in about 2-4 hours without sufficient decrease in the pressurizer pressure to cause bulk boiling. One may note that some service water from KW is available. Therefore, the loop could be cooled more quickly if necessary.

The case of complete power loss to both K areas has not yet been considered. Since such an incident would lead to loss of all recirculation pumps and the opening of all four dump valves from low-low flow trips as well as heat exchanger coolant supply loss, this case will be considered in II-B-5, "Loss of Recirculation Pumps".

The question of permitting a relaxation of the heat exchanger exit temperature high trip setting has been raised. Analysis of this process shows that the worst case with regard to the high trip setting involves the partial loss of heat exchanger cooling water such that the heat exchanger loop water exit temperature closely approaches but does not equal the heat exchanger exit temperature high trip setting. In such an instance, the maximum process tube outlet temperature following water loss would exceed its trip setting by the same number of degrees as the heat exchanger exit temperature high trip setting at the time of the incident exceeded the presently specified limit. For example, if the heat exchanger exit temperature high trip setting were increased in all cases by 15°F above the limit calculated from the present Process Standard 40.02-I, then the maximum tube outlet water temperature for the "worst case" accident would exceed the tube outlet water temperature high trip by 15°F. Therefore, no relaxation of the heat exchanger trip setting is possible since each set of test limits is based on burnout calculations for the limiting trip conditions: low flow, high tube outlet water temperatures, low system pressure.

In summary, with the possible exception of the complete K areas power loss case no damage to the loop, the fuel elements or the reactor will occur in the event of loss of cooling water to the loop primary heat exchangers as long as the heat exchanger exit temperature trip is retained in the #1 Safety Circuit at its present setting as specified in the Process Standards.

* All four loops would be affected.

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4. Coolant Loop Rupture

4.1 General

In any high-pressure flow system, one must continually guard against the occurrence of a break or rupture in the system. This is especially true in the case of a system such as a KER loop, where a line rupture could lead to a serious nuclear incident.

The most probable points of rupture in a KER loop are at the front and rear connectors or "pigtailes", although a rupture is possible anywhere in a loop. For this discussion, all ruptures will be grouped under three headings:

- a. Front-face ruptures (upstream of process tube),
- b. Rear-face ruptures (downstream of process tube), and
- c. Process tube ruptures.

Unfortunately, one cannot determine analytically at this time the flow and pressure transients following a line rupture with any more assurance of accuracy than in the case of a dump valve accident. Therefore, the three types of rupture will be discussed only qualitatively in an effort to point out real and potential problems. A quantitative discussion of this type of accident will be included in a subsequent report on the dump valve and line rupture cases.*

A line rupture can range in severity from a pin-hole leak to a complete line break. In the former case, very little coolant will be lost from the loop unless the break increases in size as time passes, which result is quite possible, depending on the cause and type of the break. Complete loss of coolant could be the result in the latter case. Since the size of the break cannot be predicted, one must treat the worst case of complete failure.

For this discussion only, a severe rupture is defined as a rupture in which the break area is significantly greater, percentagewise, than the dump valve port area.

4.2 Rear Face Ruptures

No matter how severe a rear-face rupture may be, coolant will always be available to the process tube from the loop or the process backup supply. The important questions in this accident are:

* See page 20, Paragraph 4.

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- a. Will the loop depressurize too fast, thereby leading to bulk boiling and fuel burnout before the backup coolant flow and the power reduction from rod insertion can become effective?
- b. If bulk boiling does occur, but burnout does not occur in the first seconds following the rupture, will the backup coolant pressure be sufficient to sweep steam from the process tube and restore coolant flow?

In the case of a severe rupture during high-pressure, high temperature recirculation operation, bulk boiling in the process tube almost certainly would occur. If the fuel element test were being run at near-maximum burnout design conditions, fuel element burnout might occur also, unless the depressurization were so fast that process backup flow could become effective within seconds after the rupture. However, this latter occurrence is rather unlikely.

It has been stated previously that connector failures are more probable than severe ruptures of other piping. For a connector break, the entire transient process would be similar to but more severe than a dump valve incident. If the break were less severe than a complete connector rupture such that some fluid could still flow past the break to the downstream piping, the early portions of the flow and pressure decays would be almost the same as for a dump valve incident. When the low-low flow trip points were reached, the dump valve would open, thus approximately doubling the "rupture" area. Since critical flow through the rupture would be the limiting factor at this time, the increase in flow area would be beneficial. There is also the possibility that the PS-1 pressure switch, which is used to open the dump valve when the loop pressure reaches 50 psi above process backup pressure during a fission products release and depressurization cycle, would open the dump valve during the extremely rapid low pressure transient which would occur in the first few hundredths of a second following the rupture. Opening of the dump valve at this time might introduce an unfavorable complication in the case where some fluid can reach it. No data on the character of the low pressure transient or on the response time of these switches are available. However, approximate analytical calculations of a line rupture incident, which were performed in conjunction with this study, indicate that the low pressure transient is completed within several hundredths of one second after rupture. On the other hand, if one assumes conservatively that a PS-1 switch and a rapid-response reactor Panellit gage, both of which are Bourdon tube instruments, have comparable response times, a low pressure transient of at least 1.5 - 2. seconds duration would be required to cause a PS-1 switch to register a trip and open a dump valve.

Therefore, it can be concluded that a severe rupture of a rear face crossheader or main loop pipe could lead to fuel burnout and possible reactor damage.

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Unfortunately, the second question cannot be answered without depressurization data. Therefore, for the present, one must assume that the answer is "no" for the case of severe rupture and is "yes" for the case of limited steam formation, i.e., moderate or small rupture.

A rear-face piping rupture and its potentially serious consequences are protected against by use of a procedure of static pressure testing and visual inspection of the piping at frequent intervals, which procedure is outlined in the Process Standards.

4.3 Front-Face Ruptures

The process backup water line is the only source of emergency coolant for the present loops. A small leak in the front face piping probably could be tolerated. However, if the leak became significantly larger or the original rupture were severe, the fuel elements probably would melt down in the present system, since all, or nearly all, of the coolant would be lost while the charge was near or at full power generation.

As is the case with rear-face ruptures, protection against a front-face rupture disaster is afforded by a procedure of static pressure testing and visual inspection of the piping at frequent intervals. In view of the damage which would follow a severe break, one cannot emphasize too much the importance of this testing and inspection procedure.

One may note that the design for the revised loops includes provision for an emergency rear-face coolant supply to help protect against damage from front face ruptures. Adequacy of this supply has been established by theory, (26)(27) but not by experiment.

4.4 Process Tube Ruptures

Process tube ruptures could be caused by four factors:

- a) Excessive operating pressures,
- b) A defect in the tube when new,
- c) Damage of the tube during charge-discharge operations, or
- d) Burnout of the fuel elements leading to melting of the tube.

Excessive operating pressures are prevented through use of pressure-relief devices on the pressurizer and high-system pressure alarms.

The second cause is guarded against by use of a very stringent tube testing program prior to installation⁽¹⁷⁾ in addition to the static pressure testing already mentioned. The third cause also is guarded against by pressure testing. In addition, programs have been initiated to:

- a. Develop a reliable borescoping method so that the inner surface of the tubes can be inspected periodically for evidence of damage (scratches, gouges, etc.) from charging of self-supported fuel elements.

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- b. Obtain a suitable lubricant for use during charge-discharge operation to help minimize damage from the fuel element supports.

The fourth cause can be guarded against only by attempting to prevent fuel burnout in the first place.

The effects of a tube rupture will vary depending on the location of the rupture. With regard to fuel damage, a break near the front of the tube will be as serious as a front-face rupture, while a break at the downstream end will approach a rear-face rupture in severity with regard to loop damage. In addition, it has been shown conservatively that considerable graphite stack damage, including blockage of two rows of VSR channels, could result from the surge of steam into the stack following a tube break.⁽³⁾ Therefore, a tube rupture incident has a greater potential for damage than do other types of ruptures. This greater potential underscores the necessity of the pre-installation tests of the tubes and the necessity of care in installing the tubes and servicing them (fuel charge-discharge, connector replacement, fuel rupture removal, etc.), as well as the necessity for the static pressure tests.

4.5 Summary

A small leak anywhere in the loop would not lead to serious consequences unless such a leak continued to grow rapidly in size from continued operation of the coolant system or unless the leak were in the process tube. A severe line rupture would lead to a rapid depressurization of the process tube and bulk boiling, and quite probably, to fuel burnout before the present or proposed backup coolant supplies became effective. In addition, severe graphite stack damage would result from a process tube rupture. Therefore, the static pressure test program and the tube test program are, and will continue to be, the most important methods of protecting against loop piping ruptures.

5. Loss of Recirculation Pump(s)

5.1 General

The two recirculation pumps in each loop are connected in parallel to common suction and discharge piping. When both pumps are operating, each supplies one-half of the flow. Such a situation is denoted by operating point A in Figure 4, which shows the pump characteristics and loop resistance curves for 60 gpm flow rate.

There are four pump loss cases which must be considered. These cases are listed below with an estimate (where possible) of their probabilities of occurrence:

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PUMP AND LOOP FLOW CURVES

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- a. Loss of one pump, second pump continues full-speed operation.
- b. Loss of primary (BPA) power to both pumps, emergency power assumes pumping load (1/10 yrs.)⁽⁴⁾
- c. Complete power loss (BPA and emergency) to both pumps (1/50 yrs.)⁽⁴⁾
- d. Loss of one pump, second pump is down.

5.2 Loss of One Pump, Second Pump Running

When one pump in a set drops off the line, the second pump merely exchanges head for flow and the operating point moves from A to either B or C (Figure 4) depending on whether or not the system flow control valve can adjust.* Although the system is designed to allow such a transition to take place with little or no fluctuation in loop flow, fluctuations of as much as 8-10% have been noted in extreme cases. These fluctuations are caused by sluggish response of both the check valve downstream of the stricken pump and the main flow control valve. As has been noted in II-B-1, the fuel element tests are designed so that a 20% flow reduction can be safely tolerated. Therefore, no safety hazard is involved in this case. However, one may note that the 8-10% flow decrease mentioned above can cause a "spurious" reactor scram from the differential pressure switches on the pumps (Figure 1). This problem is discussed in detail in IV-C-5, page .

5.3 Loss of Primary (BPA) Power to Both Pumps, Emergency Power Assumes Pumping Load

This case has been treated by W. J. Love.⁽¹⁸⁾ The flow decay equation used by him is of the form:

$$\frac{G}{G_0} = \frac{1}{1 + \theta/\theta_n} \quad (2)$$

Where: G = Flow rate, gpm, at time θ , sec.

G_0 = Equilibrium flow rate, gpm.

θ_n = Decay constant

*The flow control valve in Loop #1 is generally wide open at equilibrium, two-pump operation, so the system curve for this case would remain unchanged.

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Love defined Θ_n as follows:

$$\Theta_n = \frac{1}{\rho \phi G} \left[(\sigma + \rho) + \sqrt{(\sigma + \rho)^2 - 4(1 - \frac{\rho}{G})\sigma\rho} \right] \quad (3)$$

Where:

$$\sigma = \frac{\rho^2 G^2}{2g} \sum \frac{L}{d} ; \rho = \frac{e I n^2 \rho}{2g}$$

The definitions of symbols are given in Appendix A.

More recently, S. S. Jones has re-derived Equation 2 from a different approach.* He defines Θ_M as:

$$\Theta_m = \left[\frac{I n^2 N_p}{900g} + \frac{G^2 \rho}{3.24 \times 10^6 N_p^2} \sum \frac{L}{N_p d_p} \right] m^2 E / 0.32148 \rho \quad (4)$$

Again, the definitions of symbols are given on page .

The equation derived by Jones was used in this study for two reasons:

- It has been found to agree quite well with experimental data from the reactors.
- It yields more conservative results, i.e., faster flow decays, than does Love's equation.

Under the most adverse conditions, a loss of voltage in this case can occur for a period of a maximum of 1/3 second.⁽¹⁸⁾ Therefore, equations 2 and 4 were used in conjunction with the data given in Table III to calculate the flow decay which would occur in that time period. The loop operating data in Table III represent a "worst" case for this type of incident.

TABLE III

"WORST CASE" OPERATING CONDITIONS FOR BPA OUTAGE
PUMP LOSS

<u>Variable</u>	<u>Value</u>	<u>Units</u>
Tube Outlet Temperature	295	°C
Tube Inlet Temperature	195	°C
Tube ΔT	100	°C
Flow Rate	60	gpm
Tube Power	1330	kw
Flow Decrease to Boiling, % of Equilibrium Flow	84.0	%
Duration of Power Outage	0.3	seconds

*Personal communication from S.S. Jones; derivation will be published at a later date.

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The results of the calculations are given in Table IV.

TABLE IV

"TIMES TO BOIL", BPA OUTAGE PUMP LOSS

<u>Assumptions</u>	<u>"Time to Boil", Seconds</u>	<u>Symbol</u>
Ignore Fluid Travel Time in Tube and Heat Storage in Slugs	0.59	θ_1
Ignore Heat Storage in Slugs	~ 2.0	θ_1^1

The time labelled θ_1 , was obtained by assuming that, for a given reactor power and flow geometry, the difference in fluid enthalpy between the tube entrance and tube exit is inversely proportional to the flow rate. This calculation predicts no bulk boiling. Note that it ignores both the time of fluid travel through the tube and heat storage in the fuel. However, Love⁽¹⁸⁾ has derived an equation for including the effect of fluid travel time. His equation was used in conjunction with Equation 4 to calculate the time labelled θ_1^1 . From this calculation, which is still conservative, one can conclude that, under the worst operating conditions for this accident, bulk boiling would not occur in the process tube. Nor would burnout occur, since the flow will not decay below about 85% of equilibrium flow in 0.3 seconds.

5.4 Complete Power Loss to Both Pumps

Although this incident has an extremely low probability of occurrence it still must be studied to determine what damage, if any, would result from its occurrence.

A complete power loss to both pumps would cause the loop flow to decay to the low-low flow trip in 1.76 seconds, at which time the dump valve would open and a loop depressurization transient would begin. There are two differences between this case and the case of inadvertent dump valve opening:

<u>Item</u>	<u>Pump Loss</u>	<u>Dump Valve Opening</u>
Loop Flow at Time of Dump Valve Opening, % of Equilibrium Flow	64.	100.
Heat Exchanger Flow, % of Normal	~ 0	100.

Both of these differences tend to make the double pump loss case more serious than the dump valve case. In view of the fact that the two cases are so similar and involve the same analytical problems, accurate analysis of the pump loss case must await the

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completion of the depressurization test. However, the predictions concerning the adequacy of a loop to handle the dump valve case also should apply to the double pump loss case. This conclusion is based largely on the fact that although the pumps were ignored in both previous studies of the dump valve case^(13,14) and the heat exchanger was considered inoperable in one of them⁽¹⁴⁾, both authors concluded that the loop flow transient would be adequate.

5.5 Loss of One Pump, Other Pump Already Down

This case is identical to the double-pump loss case with two exceptions:

- a. Pump loss can occur from mechanical failure, so the heat exchangers need not be affected.
- b. The flow decay to the low-low trip will be slightly faster, with the trip occurring in this case in 1.22 seconds.

However, in view of the fact that tube exit temperature and pressure must be reduced to 130°C and 500 psig, respectively, during single pump operation to comply with Standards, the same conclusions concerning flow adequacy apply to this case as to the dump valve opening and double-pump loss cases.

5.6 Summary

With the possible exception of the double-pump loss case, credible pump loss incidents will not lead to reactor hazard. The excepted case is believed to be safe, but a final decision cannot be made until accurate transient depressurization data are obtained.

6. Bursting of Pressurizer Rupture Disk

6.1 Statement of Problem

Each pressurizer is equipped with two vent lines in parallel, each of which contains a rupture disk and a safety valve in series, with an open 1/2" vent line between the disk and valve. Only one line leaves the pressurizer. The dual vents are connected to this line through a three-way valve so that only one vent line is in use at any given time. With such an arrangement, the safety valve is of little value. If a rupture disk should burst, the steam from the pressurizer could flow without restraint through the 1/2" vent line and out of the stack until the three-way valve position was changed to valve in the alternate vent line.

An increase in pressurizer pressure to the disk burst-pressure would be signalled by a high-pressure alarm. Subsequent loss of fluid after bursting would cause the flow output of the loop

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make-up pumps to increase markedly, which would set off a second alarm. Thus, it is quite probable that a rupture disk accident caused by excessive pressure build-up would be detected quite soon after its occurrence even if one alarm failed to operate. However, a rupture disk can fail at normal pressurizer operating pressures from the effects of long use. In this case, if the make-up pump alarm failed to operate, no warning signals would be received by operating personnel.

Calculation of the pressure decay of a pressurizer following the bursting of a rupture disk can be simplified and can be made conservative by making the following reasonable assumptions:

- a. The steam velocity at the vent line exit to the stack can be computed by assuming that no energy is transferred from the steam to the surroundings during the expansion and flow through the vent line.
- b. Heat losses to the surroundings from the pressurizer are negligible.
- c. The automatic heaters operate at full power until the pressurizer liquid level drops to 20% of full. The heaters are then automatically turned off.

The process tube exit temperature transient following disk bursting can be determined for any given equilibrium tube exit temperature and tube ΔT_o by using the charge heat decay curve from reference 16 with the following:

$$0 \leq \theta \leq \text{Time at which reactor is scrammed} = 2 + 1 \text{ loop recirculation time in sec.} = \theta_2, T_o = T_{in,\theta} + P_{\theta} \Delta T_{t,\theta}$$

$$\theta_2 < \theta, T_o = T_o - \Delta T_{\theta t} (1 - P_{\theta} / P_o)$$

$$\theta - 1 \text{ recirc. time}$$

The reactor would be scrammed either by a system pressure or a pressurizer liquid level low trip, depending on the transients of pressure and liquid level versus time.

Curves for the system actual pressure and saturation pressure transients following the bursting of a rupture disk were computed for the following cases in Loops 2, 3, and 4, assuming a rupture disk failure at 1665 psia in each case and setting of system trips as specified in the Process Standards:

a. $T_{o,o} = 303^\circ\text{C}$

$$T_t = 20^\circ\text{C}$$

$$\text{Flow} = 27,900 \text{ lbs/hr}$$

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- b. $T_{0,o} = 200^{\circ}\text{C}$
 $T_{t,o} = 101^{\circ}\text{C}$
Flow = 27,900 lbs/hr

These curves were then used in conjunction with accepted burnout calculational procedures to determine curves of O_R versus time for the two cases. These final curves are shown as curves a and b in Figure 5.

An examination of curves a and b reveals that burnout conditions would be encountered in case b, but not in case a. The burnout situation illustrated by curve b is possible because of two factors:

- a. The depressurization of the pressurizer cannot be stopped unless the vent line valve is closed, and
- b. The Process Standards do not specify a minimum absolute pressure at which the contacts of the system pressure and boiling point suppression low trips must open, regardless of how low the tube exit temperature high trip may be set.

As an illustration, in the case depicted by curve b, the two pressure trips can be set without violating existing standards so that the reactor is not scrammed until system pressure is below 500 psia. These results also are representative of Loop 1 despite different operating limits.

It should be pointed out that case b does not represent a typical case in that the normal production test covering any given fuel charge specifies a minimum system pressure. Therefore, the situation illustrated by case b would result only in carelessness or inexperience. It is included in this report to: a) point out a potential problem, and b) provide support for the vent line revisions suggested below.

One may note that severe leaks in such places as valve to pipe connections on the pressurizer external liquid leg could lead to the same depressurization and burnout potential associated with rupture disc failure.

6.2 Proposed Solution

6.2.1 Pressure Gage with High Pressure Alarm

The first, and most desirable solution, would involve the installation in the section of piping between the rupture disc and the relief valve of a pressure gage with a suitable high pressure alarm. The valve in the 1/2" vent line would be locked in the closed position during recirculation operation. Based on consideration of the ASME B&PV Code and the zirconium process tube stress problems, the recommended system is as follows:

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RATIO OF BURNOUT-TO-ACTUAL HEAT FLUX, ϕ_R , Vs. TIME
AFTER PRESSURIZER RUPTURE DISK FAILURE, "WORST CASE"

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- a) The high pressure alarm would be set for 15 psig.
- b) Full pressure operation would be permitted following a high pressure alarm unless the pressure gage reading increased to 25 psig.
- c) If the gage reading increased to 25 psig, the loop pressure and the temperature would be reduced to 500 psia and 130°C, respectively, but recirculation operation could continue.
- d) At the first reactor shutdown following the high pressure alarm, the offending rupture disc would have to be inspected and replaced, or written approval of Process Technology Sub-Section would have to be obtained before operation above 500 psia and 130°C could be resumed. In this way, although operator action would be required to monitor the pressure gage following receipt of a high pressure alarm, operator action would not be required to prevent a system depressurization following rupture disc loss. Another important advantage of this solution is that slow disk leaks indicative of impending disk failure would be detected by a gradual buildup of pressure as noted by the gage. Therefore, actual disk failures could be prevented by early replacement.

6.2.2 Minimum System Pressure and Boiling Point Suppression Low Trip Setting

A second approach would involve incorporating a minimum system pressure trip setting in the Process Standards. Curve c represents the same case as curve b with the single important exception that a minimum value of 1400 psia was assumed for the system pressure and boiling point suppression low trip settings. The major fault of this solution with regard to rupture disk leaks lies in the fact that positive operator action (closing vent line valve) is still required. However, this approach appears to be the best solution to the problem of steam leaks elsewhere in the pressurizer equipment.

6.2.3 Orificing 1/2" Vent Line

A simple solution would be to replace the present 1/2" free vent with a much smaller one or to orifice the present line. In this way, the rate of pressurizer depressurization would be greatly reduced. Therefore, operating personnel would have a longer time in which to take corrective action before a burnout condition is reached. However, this solution has three important drawbacks;

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- a) Operator action is required to recognize the accident and halt the depressurization by closing the vent line valve.
- b) If the accident is not noted or the proper action is not taken, a burnout situation could still be reached. In addition, the time between first reaching burnout potential and reactor shutdown is lengthened because of the slower depressurization.
- c) This solution will not affect the damage potential of steam leaks at other points in the pressurizer equipment.

In summary, the best solution for the overall pressurizer steam leak problem would be the adoption of suggestions 6.2.1 and 6.2.2.

6.3 Relief Valve Operation

The preceding discussion has dealt only with cases in which a rupture disk bursts or a steam leak occurs and has not considered the cases where a safety relief valve opens. A rapid buildup of loop pressure can occur from either pressurizer liquid level controller or pressurizer heater controller failures. In these cases, an operator probably would not have time to respond to a high pressure alarm before the rupture disk would burst and the safety valve would open. Although no data on KER loop safety valve performance have been documented, testing experience at the H-1 and KER loops indicates that extremely rapid and quite large decreases in pressure can and do occur before the relief valve can reseal. For example, pressure drops as great as 1100-1200 psi in 10-15 seconds have been observed at H-1 before the valve closes. In view of the facts that minimum system pressure for burnout calculations is usually about 150 psi below equilibrium pressure and that saturation pressures in the process tube can be within 100-400 psi below equilibrium pressure, it is recommended that the relief valve in each vent line be tested and that its opening be adjusted to minimize the magnitude of pressure reduction before valve resealing.

6.4 Three-Way Valve in Main Pressurizer Vent Line

As noted above, the main pressurizer vent line contains a 3-way valve which allows the use of only one of the two rupture disk-relief valve combinations at any given time. One of the provisions of Case 1271N of the ASME B&PV Code stipulates that at least two relief valves (or disk and valve combinations) must be used. Therefore, these 3-way valves should be removed if the present pressurizer vent system is to conform to the Code.

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However, operating experience indicates that a spare relief line should be available to maintain operating efficiency. The simplest solution to this problem is the installation in a separate vent line of an additional rupture disk-relief valve combination on each side of the vent system "downstream" from the 3-way valve. In this way, two relief lines would always be available, but, at the same time, the operating flexibility afforded by the 3-way valve would be retained. In sizing and setting these additional vent systems, the potential dangers of excessive pressure reduction should be kept in mind.

C. Nuclear Incidents (Reactor Power Excursions)

A nuclear incident is defined in this report as any emergency situation which is caused by an increase of the reactor power above its equilibrium value. Such an increase could be caused by accidental withdrawal of a horizontal control rod or the accidental discharge of a poison column. The subject of nuclear incidents for the K Reactors, including causes and quantitative results is discussed elsewhere.⁽¹⁹⁾

The most severe single tube power excursion would result from the simultaneous full withdrawal of two control rods at maximum speed from the top of the reactor.⁽¹⁹⁾ Such action could result in a gross reactor power increase of 13-56%, since the shielding action of the pile would prevent the Beckman flux monitors, which are located at the bottom of the pile, from "seeing" the surge until excursions of 30-100% had been experienced by tubes near the withdrawn rod channels.⁽¹⁹⁾

A KER tube on recirculation operation would be protected by its high temperature and low boiling point suppression trips and a tube on single-pass would be protected by its high temperature trip against incurring a maximum surge. Using conservative values for rate of power rise and instrument response during the surge, values of the minimum ratio of burnout flux to actual flux, ϕ_R , were calculated for the "worst" case discussed in II-B-1 and the following operating modes:

- a. Equilibrium recirculation - normal flow and tube outlet temperature.
- b. Equilibrium recirculation - high temperature and low flow conditions.
- c. Single-pass operation - normal flow and tube temperature.

Transient flow cases were not considered since the probability of the simultaneous occurrence of a severe flow decrease and a major power surge are not considered credible. The calculations are given in Appendix C-IV and C-V. The values of ϕ_R obtained are given in Table I. From these results, it can be seen that the calculated burnout flux would be exceeded by 5.8% for the normal recirculation case and be 26.6% for the normal single-pass case. In addition, there would be considerable boiling in the effluent system for the single-pass case. Despite these results, there are three factors which should be considered in deciding on the acceptability of the postulated "worst case" test:

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- a. A considerable margin of conservatism is built into the burnout calculations.
- b. A power surge of the severity considered in these calculations has an extremely low probability of occurrence, especially in Loops 1, 2, and 3, which are not located near the top of the pile.
- c. Evidence exists⁽²⁰⁾ that burnout conditions of moderate severity can be withstood by long (30') heater rods centered in a flow channel for up to about 30 seconds before burnout occurs.

Nevertheless, these results, along with the results from Section II-B do illustrate the necessity of considering all types of possible operation of a given fuel charge before test conditions are approved.

In summary, the possibility of incurring fuel burnout in a given charge during a reactor power surge under present conditions of reactor instrumentation can and should be protected against by proper determination of operating limits for the charge prior to testing.

Methods for considerably reducing the magnitude of both the individual tube and bulk power surges from nuclear incidents have been proposed.⁽⁸⁾⁽²⁹⁾ Adoption of these methods should eliminate potential KER tube hazards from this type of accident.

III. FISSION PRODUCTS RELEASE FROM LOOP(S)

A. Surrounding Countryside

The incidents which might lead to a burnout situation and meltdown of a part or all of the charge in a KER process tube are dump valve opening, complete pump loss, a coolant loop rupture, and a severe power surge.

The first two of these incidents (dump valve opening and complete pump loss) could lead to situations where a portion of the fissionable materials would be transported through the system piping to the KE retention basins. The fraction of the total released fission materials which would be carried to the KE effluent system would depend upon the relative timing and magnitude of the charge meltdown and subsequent tube melting. It probably would be possible under selected circumstances for the tube to remain intact, and only partially plugged, in which case most of the melted charge would be swept into the effluent system or would remain in the tube. On the other hand, a fuel burnout accompanied by rapid blockage of the tube and tube melting would lead to most of the fission products being released to the reactor stack, with some leakage to the reactor building. The KE reactor effluent system is currently operated as follows:

- a. Normally, two of the three retention tanks are used in parallel, with a constant outflow from the two operating tanks to the river to balance the inflow. The third tank is maintained essentially empty.

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- b. Upon indication of fission products release to the effluent system, the total effluent flow can be routed to the third tank and then to a storage crib for retention until the radioactive material has decayed to a safe level. It takes about five minutes after confirmation of the presence of excess reactivity to valve the effluent flow to the third tank.

The amount of radioactive particulate matter (Strontium-90, Cs-137, etc.) and absorbed halogens which would reach the river after an incident of this type is dependent on the speed of diversion of the effluent stream. Although an estimate is very difficult, it is probably reasonable to assume that the increase in river activity would be fairly small.

From the foregoing, it can be concluded that fission products release outside of the reactor stack resulting from either a dump valve opening or a complete pump loss incident would be confined primarily to the release of the noble gases and a fraction of the halogens. Much of the halogens and the particulate matter should be retained in the reactor, the effluent piping and/or the retention basins. In addition, one may note that only a few central slugs might be affected by the burnout condition during dump valve or pump loss incidents if the pressure decay transient in the affected loop is fast enough to allow process water to enter and sweep out the steam before the tube becomes "vapor bound". Unfortunately, as has been noted previously, enough data is not yet available to determine if burnout and/or "vapor binding" would occur in these cases.

A meltdown following a coolant loop rupture would probably be much more serious than the above discussed cases for three reasons:

- a. In an ex-reactor loop rupture incident, the fission products can flow directly to the reactor building atmosphere. Therefore, even if the tube melted shortly after the charge burned out, a large fraction of the fission materials could be released to the reactor building. The fission cloud would then flow through the ventilating system and out the reactor stack.
- b. All cooling to the tube might be lost after the rupture. Therefore, the complete charge would be affected.
- c. Potential exists for severe damage to the graphite stack and blockage of one or two rows of VSR channels from a process tube rupture.

Any fission products release following fuel burnout after a severe power surge would follow one of two paths. If the tube remained intact, most of the fission products would pass through a section of the loop and into

* Personal communication, T. M. Hall.

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the emergency storage tanks. If the tube ruptured, a major portion of the fission products might flow into the reactor graphite stack.

The existing reactor confinement program at Hanford, which is designated as Project CGI-791 and is discussed in references 2 and 22, has as its aim the removal by a combination of water sprays and filters of 99.9% of the particulate matter and at least 50% of the halogens from the fission cloud before this cloud is released to the atmosphere.

One may note that a fog spray system, which comprises Phase I of the confinement program, is installed and manually operable. This system provides a row of nozzles which wash down the rear face atmosphere with a dense fog spray of water. Such a spray should remove much of the particulates and some of the halogens from a rear-face fission product cloud resulting from meltdown or slug fire. The overall CGI-791 project is scheduled for completion by March, 1961.

A more detailed discussion of fission products release is given in Appendix E.

B. KE Reactor Building and 1706-KE Building

Deposition of fission products inside the reactor building would be difficult to determine. However, it is reasonable to assume that deposition might be considerable for the line rupture cases under present conditions. Only the front face connector rupture case would lead to considerable building contamination for post CGI-791 conditions, since the rear face fog spray would "clean up" after a rear face incident.

The 1706-KE Building should be affected much less seriously than the KE reactor building, since the two are connected only by a piping tunnel which is not located near the path of the reactor building air flow. The components of the affected loop which are located downstream of the dump valve should remain free from contamination, since three of the incidents which could lead to burnout also cause the dump valve to open to drain, thus sealing off a major portion of the loop, while the fourth incident (severe reactor power surge) would result in the routing of the loop flow into the emergency storage tank.

IV. RELIABILITY OF AND NECESSITY FOR THE VARIOUS TRIPS IN A LOOP NO. 1 SAFETY CIRCUIT

A. General

The study of the subject of the reliability and necessity of the various trips in a loop No. 1 Safety Circuit was motivated by consideration of two factors: 1) reactor and personnel safety, and 2) operational continuity. As has been repeatedly stated by management, safety must come first. However, over-conservatism in operating a system can be

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as detrimental as lack of conservatism. Therefore, the subject herein discussed was approached with the idea in mind that the optimum condition involves a balance between conservatism and realism.

Each KER loop has twelve trips in its No. 1 Safety Circuit:

1. Process Tube Exit Temperature - High
2. Process Tube Exit Pressure - Low
3. Boiling Point Suppression (Tube Exit) - Low
4. Heat Exchanger Exit Water Temperature - High
5. Recirculation Pump Differential Pressure - Low
6. Loop Flow - High
7. Loop Flow - Low
8. Loop Flow - Low-Low
9. Process Radiation (Element Rupture) - High
10. Pressurizer Liquid Level - Low
11. Dump Valve Selector Switch - In "Recirculate" Position
12. PS-1 Pressure Switch - Low-Low (Less than Process Backup Pressure + 50 psi)

Although each trip has a specific primary protective purpose, many trips also serve to back up one or more other trips.

B. Reliability

Each trip mechanism of a safety circuit should be analyzed as it is designed to determine if it is "fail-safe", i.e., if a reactor scram will always be initiated when a scram condition exists. Such fail-safe operation can be closely approached by constructing each element of the circuit in such a way that any failure in any trip mechanism will cause a reactor trip condition to exist in the affected component. The one area of doubt is the case of jammed indicating needle on an instrument in which the needle must move either up or down a scale to a contact in order to register either a legitimate trip or a failure condition. Although this type of failure has an extremely low probability of occurrence, it has been the general policy at Hanford to require that key trips be backed up by one or more other trips. In this way, the probability of not scrambling the reactor when a scram condition exists can be reduced to insignificance.

This rigid policy on safety circuit make-up has led to many spurious reactor scrams from instrument failure. To counter this situation, and thereby improve operating continuity, the philosophy of paralleling trips, using either two or three in a matrix, has been introduced in certain instances. The system which results from this philosophy involves:

1. Two (or three) trip devices, and sometimes two (or three) sensing devices for each measured process variable or related variables in the safety circuit.

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2. The two (or three) trips connected in a parallel so that:
 - a) A failure of one during normal operation will not scram the reactor, but
 - b) a failure of one during the existence of a scram condition will not prevent a scram.
 - c) Only two of three must register trip conditions to produce a scram.

It has been the policy to require that, unless special circumstances dictate otherwise, paralleling of only two trips will be allowed only if there is also in the circuit a backup trip which will safely scram the reactor in the event the paralleled system fails.

With the notable exceptions of the process tube exit trips, the present No. 1 Safety Circuit of each loop complies with the above-stated policies and principles. Changes in some trip circuits have been proposed in order to increase operational continuity.⁽²³⁾ Where necessary, proposed changes will be discussed in the next section. However, it should be noted here that each and every proposed change must be carefully examined, with the emphasis on safety, not on operational continuity.

C. Necessity of Trips

1. Process Tube Outlet Water Temperature - High trip
2. System Pressure Low Trip in Series, with
3. Boiling Point Suppression Low Trip in Parallel.

The determination of the maximum high trip and equilibrium operating temperatures for a given test has been discussed in II-B-1, "Fuel Element Burnout". In these burnout calculations a minimum system pressure is assumed. Since the use of these calculational procedures, coupled with the required trip settings in the loop safety circuit, provides the only means of preventing fuel element burnout, these three trips are all necessary. Note that the system operates as follows:

- a) Low system pressure - boiling point suppression and system low pressure trips operate in parallel to scram reactor.
- b) High tube outlet temperature - boiling point suppression and tube outlet high temperature trips operate in parallel to scram reactor.

Thus, this set of trips illustrates the apparently successful application of the parallel trip philosophy. However, this set of trips does not have any reliable back-up, which situation constitutes a violation of accepted policy.

4. Heat Exchanger Exit Temperature - High Trip

It has been shown in II-B-3, "Loss of Coolant to Heat Exchanger", and Figure 3 that a dangerous condition would exist in a loop following loss of coolant water to the primary heat exchangers if the subject trip were

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not operable and the process tube exit trips had to be used. It also has been shown that the trip setting for the heat exchanger exit temperature could not be raised without compromising reactor safety.

Either a two of three matrix, with three independent circuits (including three separate sensing devices) or a system with four independent sensing devices feeding each of two independent circuits in each proposed loop master circuit should be installed. As is demonstrated by curve 3 in Figure 3, the process tube exit trips cannot be considered as acceptable backups to the heat exchanger trips for a heat exchanger coolant-loss accident. Therefore, use of only two paralleled trips for the heat exchanger exit temperature, which is proposed in reference 23, is unacceptable within the framework of present policy.

5. Pump Differential Pressure - Low Trip

Each of the two paralleled pumps in each loop is provided with a low differential pressure trip. The trips for each set of pumps are in parallel, thus requiring both to register a trip condition to scram the reactor. Process Standards require that these trips be set so that a decrease in differential pressure of 25% will cause the trip contacts to open. In practice, the setting of the trips usually is done while the loop is cold. From the pump characteristic curves for hot and cold flow, Figure 6, it can be seen that, if the pressure switches are set for 75% of cold flow $4\dot{p}$ at 30 gpm, then they will be set for 85% of hot flow $4\dot{p}$. If one assumes that $4\dot{p} \propto G^{2.0}$, then a decrease in loop flow to 92% of equilibrium flow would cause a trip. If the switches are initially set for 75% of hot flow, then a flow decrease to 86.7% of equilibrium flow would cause a trip. There are two cases where the pressure trips might be useful: 1) Loss of both pumps (or one pump when on single pump operation) and 2) Loss of one pump, with sticking of the check valve downstream from the stricken pump in a partially-opened position. For the first case, the decay times to the various flow and pressure trips are:

TABLE V

<u>TRIP</u>	<u>% OF EQUILIBRIUM FLOW</u>	<u>DECAY TIME, SECONDS</u>
Low $4\dot{p}$	92.0 ("cold setting")	.270
Low $4\dot{p}$	96.7 ("hot setting")	.475
Low Flow	80.0	.773
Low-Low Flow	64.0	1.743

Two things may be noted in Table V.

1. The pump Δp trips provide a trip 0.5 seconds earlier than the low flow trip when they are set with a "cold" loop and 0.3 seconds earlier than the low flow trip when they are set with a "hot" loop.

ΔP ,
psi

Notes:
1. Numbers on curves give required NPSH in PSI.
2. Fluid temperature 260°F.

PUMP HEAD vs. FLOW RATE CURVE
REQUIRED NPSH

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2. There is a delay of .97 second between opening of the low flow trip contact and of the low-low flow trip contact.

In addition, it was pointed out in II-B-5, that the only credible cause of a double pump loss would be a complete power failure at both K Areas. Such a failure would cause an immediate electrical trip of the reactors. Thus, we have a case where the reactor electric trips should cause the scram, with the pump low Δp , low flow and low-low flow trips as backup, with all trip conditions being reached 1.743 seconds after power loss. In single-pump operation, one pump could be lost without a reactor electric trip. However, Process Standards require that operating pressure and temperature not exceed 500 psig and 130°C, respectively, during one-pump operation. In this case, removal of or failure of the Δp trips would produce a situation where the scram would come essentially instantaneously from either a low or low-low flow trip if the pump were lost through shaft failure or would come .773 or 1.743 seconds after pump loss for a switching power outage or other mechanical failure.

In the case where one pump is lost from mechanical or electrical switching failure (such that KE-Reactor is not scrambled by an electrical trip) and its downstream check valve fails to close, we have the following three possibilities:

- 1) Loopflow decrease to a new rate just above the low flow trips - in this case, no danger is involved since the tests are designed to run safely at 80% of equilibrium flow.
- 2) Loopflow decrease to a new rate just above the low-low flow trip - the pump Δp will decrease to 41-45% or less of equilibrium Δp . The pump Δp trips will be the first line protection, with the low flow trip second, and the tube exit trips as further backup. If the pump loss is from a shaft break, the first two trips will be reached virtually instantaneously, while the third trip will be reached in approximately 1.6 - 2.8 seconds, depending on equilibrium tube power. If the pump loss is from any other cause, the pump Δp trip will come first, but the difference in timing between Δp trip and low flow trip will be small (i.e., ~ 0.5 seconds). The third trip still will be reached in approximately 1.6 - 2.8 seconds.
- 3) Flow decrease below low-low flow trip - the pump Δp trip contacts will open first, then the low flow, then the low-low flow, all within several tenths of a second after flow reversal in the stricken pump. The tube exit trip conditions will be reached in less than 1.6 - 2.8 seconds, depending on equilibrium tube power.

A summary of the foregoing discussion is presented in Table VI. From this summary, one can see that failure of the Δp trips to operate when needed will not lead to excessive delays in scrambling the reactor. Complete removal of the Δp trips from the No. 1 Safety Circuit would mean that failure of the low flow trip to operate in Case 4, Table VI, could lead

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TABLE VI

SUMMARY OF TRIP TIMES FOLLOWING PUMP LOSS INCIDENTS

<u>Case</u>	<u>No. of Pumps Operating</u>	<u>Cause of Flow Loss</u>	<u>Possible Trips</u>	<u>Time to Trip, Seconds*</u>
1	2	Power Outage	Reactor Electric Pump Δ p	0. (0.27 ("cold setting") (0.47 ("hot setting")
			Low Flow	0.773
			Low-Low Flow	1.743
2	1	Power Outage or any mechanical failure not involving immediate, total pump loss.	Reactor Electric Pump Δ p	0. (.188 ("cold") (.332 ("hot")
			Low Flow	.542
			Low-Low Flow	1.22
3	1	Shaft break or any other immediate, total pump loss cause.	Pump Δ p	0.
			Low Flow	0.
			Low-Low Flow	0.
4	2	Loss of one pump with sticking of its downstream check valve.		
		a) Flow reduction to 80.1%	Pump Δ p (no trip needed)	(0.27 ("cold") (0.475 ("hot")
		b) Flow reduction to 64.1%	Pump Δ p	0.27 ("cold") 0.475 ("hot")
			Low Flow	0.773
			Exit Trips	1.6-2.8
		c) Flow reduction to less than 64%	Pump Δ p	(0.275 ("cold") (0.475 ("hot")
			Low Flow	0.773
			Low-Low Flow	1.743
			Tube Exit Trips	1.6-2.8

* Times to trip for tube exit trips assume no previous trip has functioned.

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to a delay of as much as 2.8 seconds before a scram. Such a delay probably would not lead to fuel burnout. However, to maintain reactor safety and improve operating continuity, it is recommended that the following possible courses of action be investigated:

- a) The pump Δp_p and the low flow trips could be arranged in a matrix such that:
 - i) One Δp_p trip plus one low flow trip will cause a scram.
 - ii) Two Δp_p trips only will not cause a scram.
- b) The required Δp_p trip setting could be decreased to about 60% of equilibrium Δp_p at cold conditions, so that a Δp_p -caused scram would be nearly coincident with a low flow scram.

6. Loop Flow - High Trip

All three of the loop flow trips receive their signals from an orifice which is located near the front-face of the reactor, downstream from the process backup water inlet. The purpose of the high flow trip is to protect the loop against a line rupture downstream from the orifice.* The process tube exit trips serve as the backup for the high flow trip. The high flow trip contacts are set to open if loop flow increases 15% or more above equilibrium during single-pass operation or to 68 gpm or more for recirculation operation.

It has been proposed that operating personnel be allowed to bypass the high flow trip during single-pass operation.⁽²⁴⁾ This proposal was motivated by the occurrence of a reactor scram as a result of a manual valving error while the loop flow was being adjusted during single-pass operation. The argument has been advanced that the bypassing of this trip would not endanger the reactor or loop in question since the line pressure during single-pass operation is only 200-400 psig while the loop piping is normally cold (20°C) static-pressure tested at 3000 psig for Loop #1 and 4000 psig for Loops #2, 3 and 4. This argument may be quite true. However, very little is known either about the consequences of a line rupture during single-pass operation or the possible course a rupture might take. Would the rupture begin as a small leak and grow slowly, with the possibility that a high flow trip would scram the reactor well in advance of a tube exit trip, or would the two trips come almost simultaneously? In addition, the problems affecting operational continuity seem to be largely procedural in nature. Therefore, it is recommended that no changes be made with regard to the high flow trip until rupture information becomes available through various tests now being planned or run; namely, the NPR tube rupture tests, and the proposed KER depressurization and NPR

* As was noted on Page 27, the most probable points of rupture lie between the front and rear-face crossheaders.

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simulated loop rupture tests. All information from these tests should be available by the winter of 1961, at which time the subject should be re-examined. In the meantime, operating personnel should be made more aware of the consequences of their actions.

7. Loop Flow - Low Trip

The low flow trip is set to scram the reactor if the loop flow decreases below 80% of the equilibrium rate. It has been noted in II-B-1, "Fuel Element Burnout", that the fuel element tests are designed for no burnout at the low flow-high tube exit temperature-minimum system pressure operating conditions. Therefore, in many tests, operating at flow rates below 80% of equilibrium flow could very easily lead to burnout. Allowance of a 20% reduction in flow without scrambling the reactor, coupled with the allowable 15% increase to the high flow trip, provides a wide enough band around equilibrium flow to eliminate scrams of the reactor from the flow trips due to normal, minor fluctuations. Therefore, the low flow trip as installed and operated is a primary and essential protective device.

8. Loop Flow - Low-Low Trip

The low-low flow trip contacts are set to open if the loop flow decreases to 64% or less of equilibrium flow. The low-low flow trip performs three important functions: 1) it backs up the low flow trip in the event of a major flow loss (loss of both pumps, serious sticking of pump check valve after single pump loss, line rupture upstream of flow orifice, line plugging), 2) it opens the dump valve to permit rapid loop depressurization so process backup flow can enter the loop, and 3) it backs up the process backup valve trip to insure that that valve is open and process water can flow through the tube on demand. This trip is obviously necessary especially in that it provides the only automatic way of opening the dump valve to start a rapid loop depressurization to emergency flow following a recirculation flow loss.

9. Process Radiation (Fuel Element Rupture) - High Trip

One of the most probable, and potentially dangerous, incidents is the rupture of a fuel element, with the accompanying release of fission products to the loop. Apparently, many ruptures start as small breaks in the fuel element cladding. If the reactor continues to operate, the rupture opening grows in size and the element may tend to swell, thus blocking flow and presenting the very real possibility that the process tube may overheat and

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rupture in the region of the swollen element. Thus, the possible consequences of a severe rupture include gross loop contamination, extra personnel exposure, graphite stack damage, loss of production from KE Reactor, and loss of a process tube. Therefore, the loops were equipped with high radiation trips in an attempt to provide for reactor shutdown and diversion of contaminated coolant to a special tank before a ruptured element can cause serious damage.

It has been suggested that parallel radiation trips be installed in the loops to improve operating continuity.⁽²³⁾ In doing this, one must realize that there is no definitely safe backup for the radiation trip. Therefore, the principle stated in section 7.1.2, page 13, of Reference (23) must be followed:

"At least two independent devices shall be required in all coincident trip matrices. Factors to be included for selection of actual number of devices are importance of component in circuit and reliability of devices."

10. Pressurizer Liquid Level - Low Trip

The pressurizer maintains the loop pressure above the loop saturation pressure in order to prevent bulk boiling under normal operating conditions. In addition, the pressurizer performs another distinct, but not independent, function. Its pressure maintains an adequate net positive suction head at the inlets to the recirculation pumps. Without any details, it will suffice here to say that the net positive suction head at a pump inlet, abbreviated as NPSH, is approximately equal to the sum of the system static pressure and the kinetic head of the fluid minus the fluid vapor pressure, with all heads or pressures being measured at the pump inlet. From this definition, one can see that a decrease of system pressure to system saturation pressure at the pump inlet reduces the NPSH to approximately zero. The pump curves in Figure 6, indicate the relation between flow and pressure as a function of NPSH, with pump speed held constant. From these curves, it is obvious that a large decrease in NPSH must be accompanied by severe flow reductions.

Thus, the consequences of complete loss of liquid from the pressurizer during high-pressure operation would be two-fold:

- 1) General bulk boiling in the loop.
- 2) Serious reduction of loop flow rate.

Further, since the pressurizer heaters would be exposed to a steam atmosphere, they would probably melt unless turned off. Note, though, that turning off the heaters would tend to further depressurize the loop and cause a continued outflow from the pressurizer.

For the above reasons, the pressurizer liquid level indicator is provided with a trip such that the trip contacts will open and scram the reactor if liquid level is below 20% full. In addition, the heaters are automatically turned off since they are exposed, at least in part, to a steam

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atmosphere when the liquid level drops below about 15%. In normal operation, the liquid level is maintained between 50 - 70% of full to provide for loop water contraction during periods of rapidly decreasing temperature. Three possible cases where this trip provides protection to the reactor are:

- 1) Failure of the heat exchanger by-pass valve such that all loop flow goes through the heat exchanger.
- 2) A slow leak anywhere in the system which causes more water to be lost from the system than can be replaced by the make-up pumps.
- 3) Bursting of a pressurizer ruptured disk.

All of these cases are not only possible, but quite probable, In the first two cases, a low liquid level trip probably would provide the first automatic scram of the reactor. As a matter of fact, this trip would be the only automatic one until the loop pressure reached the point where the tube exit pressure low-boiling point suppression low trip combination would function. However, at this point, it might already be too late, especially if an initially small leak grew in magnitude with time. Even assuming that the tube exit trips theoretically would function in time, elimination of the low level trip from the safety circuit would mean that failure of the tube exit trips to function would leave the loop without any means of scrambling the reactor until the flow stoppage-bulk boiling situation was experienced. Therefore, liquid low level trip is a necessary component of the No. 1 Safety Circuit.

However, it would be possible to parallel three independent liquid level sensing and trip systems, thus eliminating the possibility of spurious scrams from liquid-level instrument failure. It has been noted that a backup to this system from the tube exit trips may exist. In determining the adequacy of the tube exit trips as a backup, much would depend on the dynamics of the pressurizer-loop system, including such things as pressurizer heat losses to the atmosphere, rate of loop liquid contraction, size of leak, etc., for any given case. With our present knowledge, the only safe course is to assume that no backup to the liquid level trips exists. Therefore, although the parallel trip system is acceptable in principle, as in the case of the radiation and heat exchanger trips, the principle from reference (23) must be followed.*

11. Dump Valve Selector Switch - In "Recirculate" Position
12. PS-1 Pressure Switch - Loop Low-Low

A manual dump valve position selector switch was provided in each loop control system so that a loop can be operated on:

* See Page 53, Line 13.

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- a) Single-pass flow with the dump valve locked open to "Dump",
- b) recirculation flow at pressures above the sum of the process backup water pressure plus 50 psi with the switch in the "Automatic" position, so that the valve position is determined by the low-low flow and low-low pressure switches, or
- c) recirculation flow at pressures below the sum in b with the switch in the "Recirculation" position, so that the valve position is determined by the low-low flow switch only.

In order to insure that the selector switch would be in the "Automatic" position, a trip was incorporated into the safety circuit which will scram the reactor if the switch is moved to the "Recirculate" position during high pressure recirculation operation. This trip can be by-passed manually for low pressure operation.

The low-low pressure switch (PS-1) opens the dump and process backup water valves during the automatic depressurization following fuel element rupture. If the low-low pressure switch (PS-1) were removed from the No. 1 Safety Circuit, then only the low-low flow switch would be available to open the dump valve automatically in an emergency. Situations can be postulated where the PS-1 switch might be useful in opening the dump valve and preventing reactor damage: a) line rupture upstream from the flow orifice, with a coincident failure of the low-low flow switch, b) pressurizer leak or blown pressurizer rupture disk, with coincident failure of low-low flow switch and of personnel to react properly. In addition, this switch provides automatic protection against loop contamination following a fuel element rupture.

In view of the possible reactor safety hazards and the fact that the only scram resulting from turning of the selector switch to "Recirculate" during high pressure operation was caused by a human error, it is recommended that the dump valve selector switch be electrically interlocked to the proper by-pass switch (By-Pass #111) so that the selector switch will not move from "Automatic" to "Recirculate" unless the selector switch is first by-passed. The procedures governing by-passing will remain as now stated in the Process Standards.

D. Compliance of Pressure Sensing System with ASME Code

The recent ruling⁽⁷⁾ of the ASME B&PV Code Committee previously referred to in II-B-6 stipulates that a pressurized water reactor system must have two independent pressure sensing systems which do not depend on the same source of external energy. The KER loops meet the independent sensing system requirement in that the pressurizer pressure sensing system and the tube exit pressure and boiling point

suppression system are independent of each other. However, both of these loop systems rely on the same external electric power supply. A recent document⁽²³⁾ recommends separate power supplies for each circuit of a proposed paralleled safety circuit net for the loops. One power supply would come from 165-KE and one from 165-KW. The design change for these revisions must include provision for powering one pressure sensing system from each power source in order to comply with the ASME Code.

V. COMPLIANCE OF LOOPS WITH COOLANT SYSTEM CRITERIA

A. General

The coolant system safety criteria⁽⁴⁾ which were adopted recently for Hanford reactors were originally conceived and documented for application to the production reactors. However, since the KER Loops are a part of KE-Reactor and are, in essence, little reactors, it is reasonable and necessary to require that operation of these loops shall conform to the criteria. These criteria are given in Appendix D for the convenience of the reader.

The conformance of the loops to these criteria will be discussed in two parts: single-pass operation and recirculation operation.

B. Single-Pass Operation

During normal single-pass operation, a 1706-KER tube is nothing more than a specially charged and piped KE Reactor process tube. Since KE Reactor has been found to conform to the criteria with exceptions as noted in reference 4, one need only consider the effects of the differences between a KE tube and a KER tube.

The principal differences of interest to this subject between the tubes in question are:

1. The KER tubes are usually loaded with seven-rod cluster and tube-and-tube elements, while KE tubes contain I & E slugs. These differences affect the pressure drop and burnout calculations.
2. The KER tubes are connected through high pressure nozzles and a dump valve directly to the KE effluent line, while a KE tube empties into a crossheader to riser to downcomer effluent system. Therefore, a KER tube has no rear face coolant supply in the event of a front face line rupture.

From the discussions of burnout and coolant loop line rupture given in II-B-2 and II-B-5, it can be concluded that:

1. If the single-pass operation is carried out within the framework of the accepted burnout calculational procedures and limits, the primary coolant supply will conform to the criteria, except as noted under item 3 below.

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2. The secondary and last-ditch systems conform to the criteria to the same extent as does KE Reactor, except as noted under item 3 below.
3. The loops are not now protected against a sizable line rupture in the front face piping since all three coolant systems (primary, secondary, and last-ditch) feed the tube through this piping. In addition, fuel burnout can result from a severe loop rupture in the rear face piping.

Therefore, criteria b and d, concerning independence of the systems and single component failures, are not upheld. As has been noted in II-B-5 the scope for the revised KER loops includes a rear face coolant supply for each tube. Proper design of this system will greatly reduce, but will not eliminate, the line rupture problem. Protection during operation can be obtained only by a systematic program of loop pressure testing and tube inspection and testing. Fortunately, the probability of incurring a damaging line rupture during single-pass operation is small since the loops are constructed of high-pressure piping, (cold test pressure is 4000 psig), while process water pressure is never more than about 480 psig.

C. Recirculation Operation

In recirculation operation, the primary cooling system consists of the main loop piping, heat exchanger and two paralleled pumps. This system is adequate to handle all normal cooling loads for the simple reason that all fuel element tests are designed with this factor in mind. However, as has been pointed out in Section II, the adequacy of the primary system to handle transients following various incidents is open to question in four cases: opening of a dump valve during reactor operation, complete pump loss, severe power surge, and coolant loop rupture. Unfortunately, none of these cases can be accurately treated, although it is reasonable to conclude tentatively that the first three incidents can be handled safely while a severe loop rupture will lead to serious damage. However, one should be cautioned against acting on this conclusion until more information becomes available. One may note that during any transient involving the opening of a dump valve, the pressurizer and its contents automatically become a part of the primary system.

The secondary and last-ditch cooling systems for recirculation operation coincide with the primary, secondary and last-ditch systems for single-pass operation and KE Reactor operation. Therefore, the secondary and last-ditch systems for recirculation operation conform with the criteria except for the case of a front face line rupture. In this case, the comments for single-pass operation apply here, except that operating pressures for recirculation operation are 1250 psig for Loop #1 and 1650 psig for Loops #2, 3, and 4.

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APPENDIX A

NOMENCLATURE

Note: Only terms not defined in text are listed here.

Roman Letters

A	Surface Area, ft^2
D	Diameter, ft.
E	Pump Efficiency, $\%/100$
e	Overall efficiency of Pump and Pump Motor, $\%/100$
F	Flow rate, lbs-M/hr
G	Flow rate, gal/min
g	Gravity constant, ft/sec^2
I	Moment of inertia-pump rotor, shaft and flywheel, lbs-F - ft^2
L	Length, ft.
n	Pump speed, rpm
N_p	Number of pumps in parallel
N_p	Number of pipes in parallel
Δp	Pressure drop, psi
P	Absolute pressure, psia
Q	Flow rate, ft^3/sec
S	Cross-sectional area, ft^2
T	Temperature, $^{\circ}\text{C}$ unless otherwise noted
ΔT_{hl}	Coolant temperature difference, process tube inlet temperature minus heat exchanger exit temperature.
ΔT_t	Temperature difference across process tube, $^{\circ}\text{C}$ unless otherwise noted
V	Fluid velocity, ft/sec

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(APPENDIX A Cont'd)

Greek Letters

- θ Time, seconds
- ρ Fluid density, lbs-M/ft³
- ϕ Heat flux, BTU/hr-ft²
- ϕ_p Pump constant in Equation 3, =0.20
- ϕ_R Ratio of burnout-to-actual heat flux

General Subscripts

- BO Burnout
- f Friction
- HT Tube exit high trip temperature condition
- L Local condition at critical heat flux point
- LF Tube exit low trip flow condition
- l Loop (less pumps)
- O Process tube outlet
- θ Equilibrium condition
- P "per pump"
- p "per pipe in parallel"
- S Saturation
- t Process tube

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APPENDIX B

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APPENDIX C

SAMPLE BURNOUT CALCULATIONS

The following sample fuel element test is postulated, based on a seven-rod cluster fuel element design and a maximum expected KER tube power.*

TABLE C-1

DATA FOR SAMPLE BURNOUT CALCULATIONS

<u>Variable</u>	<u>Value</u>	<u>Units</u>
Tube Power	1400	kw
Critical Heat Flux	0.53	BTU/hr
Equilibrium Tube Outlet Temperature, $T_{O,o}$	200.	°C
Equilibrium Tube ΔT	101.	°C
Equilibrium Critical Local Temperature, T_L	250.5	°C
Equilibrium Flow Rate, F	25,840	lbs/hr
Minimum System Pressure, P_M	1500	psig
Flow System Geometry:		
Fuel Rod Diameter, D_F	0.593	inches
Tube Diameter, D_T	2.028	inches
Flow Area, A_F	0.009	ft ²
Inner Channel Area, A_I	0.00033	ft ²
Outer Channel Area, A_O	0.00117	ft ²
Hydraulic Diameter, D_e	0.0225	ft
Heated Perimeter, D_1	0.346	ft
Flow per Inner Channel	0.0308 F_T	lbs/hr

* All symbols not defined in this Appendix are defined in Appendix A.

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Appendix C Cont'd

I. Equilibrium Recirculation Operation

In practice, several empirical correlations would be used to determine the proper operating conditions. For illustrative purposes, only one correlation will be used.

Bernath(3) proposed that the following empirical correlation would represent considerable burnout data over a wide range of operating conditions to an accuracy of $\pm 15\%$:

$$\phi'_{BO,L} = \left(\frac{Q}{A} \right)'_{BO} = \left[5710 \left(\frac{P}{D_L} \right)^{2.6} + 48 \frac{V}{D_L} \right] \left[57 \ln P - 54 \left(\frac{P}{AUS} \right) - \frac{V}{D_L} \right]^{0.6} - T \quad (C-1)$$

Where: $\left(\frac{Q}{A} \right)'_{BO} = \text{Pcu/hr-ft}^2$

The burnout heat flux determined from Equation C-1 is usually multiplied by 0.65 before comparison with the postulated operating flux. The factor of 0.65 insures that the calculated burnout heat flux will be less than at least 98% of all experimental values for these conditions. Multiplying Equation C-1 by 0.65 and converting the flux to BTU/hr gives:

$$\phi_{BO,L} = 1.17 \phi'_{BO,L} \quad (C-2)$$

Substituting the data for the postulated case in Equation C-2, one finds that the following is true:

1. Normal Tube Outlet Temperature

$$\phi_{BO,L} = 0.858 \times 10^6 \frac{\text{BTU}}{\text{hr-ft}^2} \quad (C-3)$$

$$\phi_R = 1.62 \quad (C-4)$$

2. High Trip Tube Outlet Temperature, Low Trip Flow Rate

$$\phi_{HT,LF,L} = 0.536 \times 10^6 \frac{\text{BTU}}{\text{hr-ft}^2} \quad (C-5)$$

$$\phi_R = 1.012 \quad (C-6)$$

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These results show that the postulated test meet the criteria given on page 15. However, in practice, the tube outlet temperature limits would be lowered so that the ratio of burnout flux to critical flux for the high temperature, low flow case would be at least 1.15 or greater, unless some strong reason existed for decreasing this ratio.

II. Equilibrium Single-Pass Operation

Assume that:

$$T_{in} = 5^{\circ}\text{C}$$

$$F_{\bullet} = 70 \text{ gpm (measured at } 100^{\circ}\text{C)}$$

Under these conditions:

$$T_0 = 5 + 1 \left(\frac{60}{70}\right) 101 = 91.5^{\circ}\text{C}$$

$$T_L = 5 + 3/2 \left(\frac{60}{70}\right) 101 = 135^{\circ}\text{C}$$

$$P_{S,L} \approx 45 \text{ psia}$$

$$P_L \approx 280 \text{ psia}$$

Therefore, the Process Standards requirements for single-pass operation, which state that the tube outlet temperature must be less than 100°C and the minimum difference between the saturation and actual pressures in the tube must be such as to provide 25°C boiling point suppression, are met.

Substitution of these data in Equation C-2 gives:

$$\phi_{BO,L} = 1.168 \times 10^6 \frac{\text{BTU}}{\text{hr-ft}^2}$$

$$\phi_R = 2.205$$

One may note that the margin of safety for equilibrium single-pass operation is much greater than for equilibrium recirculation operation.

III. Transient Single-Pass Operation

The "worst case" transient involves an instantaneous 20% flow decrease, with an effective reactor scram occurring some 3 to 4 seconds later.

For this case:

$$T_0 = 113^{\circ}\text{C}$$

$$T_L = 167^{\circ}\text{C}$$

$$P_{SL} \approx 108 \text{ psia}$$

$$P_L \approx 185 \text{ psia}$$

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$$\phi_{BO,L} = 0.541 \times 10^6 \frac{\text{BTU}}{\text{hr-ft}^2}$$

(from Equation C-2)

$$\phi_R = 1.022$$

IV. Power Surge - Equilibrium Recirculation Operation

Assume we have the same element as in the previous cases, but, in the interest of examining the worst case, assume that:

$$\left(\frac{\phi_{BO,L}}{\phi_L} \right)_{HT,LF} = 1.0$$

This assumption requires that

$$\phi_{HT,LF} \times 10^{-6} = 0.536 \text{ instead of } 0.530.$$

A. Normal Operation (i.e., $T_{O,HT} - T_{O,\theta} = 15^\circ\text{C}$)

The most severe single tube power excursion would result from the simultaneous full withdrawal of two control rods at maximum speed from the top of the reactor⁽¹⁹⁾. Naturally, such an incident would have its maximum effect on the closest surrounding tubes. The #4 Loop process tube 4268 has a reasonable chance of experiencing such an incident. Data from reference 14 show that the maximum rate of rise of tube power (given as percent of equilibrium power per second) at the outlet temperature trip point of interest in this calculation is approximately 1.5 %/second. Therefore, assuming an effective scram time for the reactor of 5.0 seconds (includes time lag in instruments and drop time of rods) after the trip condition is reached and neglecting the small changes in fluid properties, fluid travel time in the tube and fuel heat storage, the maximum tube ΔT_t is:

$$\Delta T = 101 + 15 + (.015)(5.0)(101) = 123.6^\circ\text{C}$$

Therefore:

$$T_L = 99 + 1.5 (123.6) = 284.5^\circ\text{C}$$

And

$$\phi_{BO,L} \times 10^{-6} = 0.552$$

(from Equation C-2)

$$\phi_L \times 10^{-6} = 0.639$$

So

$$\phi_R = 0.863$$

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B. High Trip Tube Outlet Temperature, Low Trip Flow Rate

In this case, trip conditions will be exceeded immediately. Again, assuming 5.0 seconds for an effective scram, one finds from reference 14 that:

$$T_{t,HT,LF} = 116(1+0.016) = 117.8^{\circ}\text{C}$$

So:

$$T_{L,HT,LF} = 99 + 1.5(117.8) = 275.8^{\circ}\text{C}$$

$$\phi_{L,HT,LF,BO} \times 10^{-6} = 0.514$$

$$\phi_{L,HT,LF} \times 10^{-6} = 0.536(1.016) = 0.544 \quad (\text{from Equation C-2})$$

$$\phi_R = 0.947$$

V. Power Surge - Equilibrium Single-Pass Operation

Using the conditions for C-II and assuming 5 seconds from trip to effective scram, one finds that the maximum power surge will give:

$$\Delta T_t = 86.5 + 8.5 + 2.2 = 102.2^{\circ}\text{C}$$

$$T_0 = 107.2^{\circ}\text{C}$$

$$T_L = 5 + 3/2 (102.2) = 158.3^{\circ}\text{C}$$

$$\phi_{BO} \times 10^{-6} = 0.494 \quad (\text{from Equation C-2})$$

$$\phi_L \times 10^{-6} = 0.53 (1.18) = .626$$

$$\phi_R = 0.79$$

APPENDIX D*SAFETY CRITERIA1. Nuclear Control1.1. Introduction

The following "speed of control" and "total control capacity" criteria result from extensive studies, and have been adopted at Hanford. The presently adopted interpretations of the general statements of the criteria have been discussed with the ACRS and the AEC and found to be acceptable bases for safety analyses.

1.2. Speed of Control

The general statement of the speed of control criteria is:

The design and/or operation of the reactor shall be such that in the event of a power excursion resulting from any credible combination of events including complete and permanent loss of coolant, the response of the reactor safety control system shall be sufficient to insure that no energy state will develop, while the reactor is super-critical during the excursion, which would significantly increase the consequences of the accident.

Specifically, this is presently interpreted as requiring that the surface temperature of the fuel element shall not reach the melting point of uranium during the excursion.

1.3. Total Control

The general statement of the total control criteria is:

The design and/or operation of the reactor shall be such that in the event of any credible combination of events, including complete and permanent loss of coolant, the total control strength shall be sufficient to insure that no reactivity state will develop which would significantly increase the consequences of the accident.

The presently adopted interpretation of this general criteria specifies that:

A prompt critical state is never permitted. Criticality short of prompt is allowed in the reactor during the meltdown process only after a time when it is obvious that emergency procedures to save the reactor from complete meltdown are no longer possible. In the case of the present reactors, with aluminum process tubes, the time at which the reactor complex has reached the melting point of aluminum is considered the time after which a super-critical state can be allowed.

*This Appendix is a direct quotation from reference 4.

2. Reactor Cooling System Reliability

2.1. Introduction

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

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

2.2. Bases

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

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

The ultimate reliability of a system or series of systems has not yet been expressed quantitatively in a satisfactory manner nor has an acceptable level of cooling system reliability been adequately defined. On the other hand, the mandate, "the system shall not fail" as a basis or criterion is equivalent to requiring that the system not be operated. 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.

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

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

2.3. Criteria

The normal reactor cooling system requirements shall be:

2.3.1. 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 back-up 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.

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

c. Last Ditch System

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

- 2.3.2. An independent system is defined as one with power sources and pumping systems such that failure or cause of failure in one cannot induce failure of the alternate systems.
- 2.3.3. 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.
- 2.3.4. 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.
- 2.3.5. All engineering, construction, operation and maintenance shall meet or exceed requirements specified by applicable codes and standards.

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APPENDIX E

FISSION PRODUCTS RELEASE CALCULATIONS AND DISCUSSION

Healy⁽²¹⁾ has published calculations which show the effects on the surrounding country of the fission products release following the meltdown of a single tube in a 2000 MW reactor. The estimated maximum* KER tube conditions and the operating conditions for Healy's average tube are shown in Table E-I. From these data and Figure 2 of Healy's report, which is reproduced as Figure E-1, one can construct the curves shown in Figure E-2, which are the fission products decay curves for the maximum KER tube and Healy's tube. These curves show that use of Healy's results for effects of a single tube release to describe the maximum KER tube release will yield conservative results for considerations of personal damage (lethality) but will yield nonconservative results for property damage, since personal damage is assumed to result from exposure to the initial radioactive cloud, while property damage is based on contamination levels five years after the release incident. However, since Healy's results indicate order of magnitude effects only, it is felt that refinement of his calculations to fit the exact KER maximum tube release is unwarranted. Therefore, the following conclusions reached by Healy are considered to be applicable to a release from a maximum KER tube under present confinement:

Personal Damage

- "a. Lethalities would not be expected at distances from the reactor as great as one mile.
- b. In the neutral or unstable case**, exposures to external gamma would be less than one roentgen at distances of one mile.
- c. In the inversion case**, external exposures as high as one roentgen could occur at distances on the order of ten miles."

TABLE E-I

DATA ON TUBE CHARGES USED IN FISSION PRODUCT RELEASE CALCULATIONS

<u>Variable</u>	<u>Maximum KER Tube*</u>	<u>2000 MW Reactor Tube⁽¹⁷⁾</u>
Power, kw	1400	2000
Exposure, days	536	200
Uranium Weight, lbs.	300	270
Exposure, MWD/ton	5000	2960

*Maximum with regard to exposure in MWD/ton.

* Maximum from the point of view of exposure in MWD/ton.

** Conclusion assumes this type of stability of the atmosphere exists during entire period of release.

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FIGURE E-1

FISSION PRODUCTS DECAY CURVES
FOR VARIOUS IRRADIATION TIMES

Notes:

1. Time on curve refers to element exposure time.
2. The Figure is reproduced from Reference 17.

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FIGURE E-2
FISSION PRODUCTS DECAY CURVES

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Appendix E Cont'd

Property Damage

The areas affected are in the vicinity of the reactor. The maximum distances and estimated areas for the various degrees of damage are listed in Tables E-II and E-III for stable and unstable atmospheric conditions for present confinement capability. Elevated release means release from the reactor stack which probably would be the case with all material except that which goes to the retention basin.

TABLE E-II

MAXIMUM DISTANCES
(Miles)

	<u>Ground Release</u>		<u>Elevated Release</u>	
	<u>Stable</u>	<u>Unstable</u>	<u>Stable</u>	<u>Unstable</u>
Long Evacuation	3-6	<1	<1	<1
Temporary Evacuation	10-20	1-2	<1	1-3
Crop Damage*				
Particulates	50-80	5-10	0-40	4-10
Halogens	5-50	60-100	<200	70-200

TABLE E-III

ESTIMATED AREAS
(Square Miles)

	<u>Ground Release</u>		<u>Elevated Release</u>	
	<u>Stable</u>	<u>Unstable</u>	<u>Stable</u>	<u>Unstable</u>
Long Evacuation	1-2	<1	<1	<1
Temporary Evacuation	4-12	<1	<1	<1
Crop Damage*				
Particulates	60-80	1-4	0-10	2-3
Halogens	3-70	<200	<200	<200

*The distances of crop damage are listed separately for the particulate and halogen damages because of the different character of the contamination.

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Appendix E Cont'd

The existing reactor confinement program at Hanford, which is designated as Project CGI-791 and is discussed in references 2 and 22, has as its aim the removal by a combination of water sprays and filters of 99.9% of the particulate matter and at least 50% of the halogens from the fission cloud before this cloud is released to the atmosphere. Healy⁽²¹⁾ has concluded that

"Although the containment problems have not been fully examined as to their influences on the course of the incident, it is indicated that the (land) areas affected will be reduced about in proportion to the degree of containment obtained."

Using this statement as a rule of thumb, one can arrive at the rough estimates of distances and areas affected for post CGI-791 confinement which are given in Tables E-IV and E-V.

No account has been taken of deposition of fission products inside the reactor building in arriving at the figures in Tables E-II to E-V.

TABLE E-IV

MAXIMUM DISTANCES - POST CGI-791
(Miles)

	<u>Ground Release</u>		<u>Elevated Release</u>	
	<u>Stable</u>	<u>Unstable</u>	<u>Stable</u>	<u>Unstable</u>
Long Evacuation	0	0	0	0
Temporary Evacuation	1-2	<1	<1	<1
Crop Damage				
Particulates	3-5	<1	0-4	<1
Halogens	4-40	30-50	<160	30-100

TABLE E-V

ESTIMATED AREAS - POST CGI-791
(Square Miles)

	<u>Ground Release</u>		<u>Elevated Release</u>	
	<u>Stable</u>	<u>Unstable</u>	<u>Stable</u>	<u>Unstable</u>
Long Evacuation	0	0	0	0
Temporary Evacuation	1-4	<1	<1	<1
Crop Damage				
Particulates	6-8	<1	<1	<1
Halogens	1-30	>20	>100	>20

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