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HAZARDS SUMMARY REPORT — PROJECTS CG-558 AND CG-600 REACTOR PLANT MODIFICATIONS VOLUME 1

BY

R. E. TRUMBLE

IRRADIATION PROCESSING DEPARTMENT

MARCH 6, 1957

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REACTOR PLANT MODIFICATIONS
VOLUME 1

By

R. E. Trumble

Process and Reactor Development
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ABSTRACT

VOLUME 1

It is believed that the over-all safety of reactor operation will be improved upon completion of projects CG-558, 600, even at the higher power levels anticipated. Installation of sub-critical monitoring instruments concurrent with these projects is a factor in this conclusion. Higher power levels will not of themselves increase the probability of a disaster initiating event; however, higher power levels will reduce the time available for remedial action and will increase the severity of the consequences of disaster.

Loss of process cooling water will precede or accompany a reactor disaster. A reactor containing a normal inventory of fission products will surely be destroyed, with release of some fission products, if cooling water is lost at operational power levels or within hours after shutting down the reactor from operational levels. A power excursion along with the water loss, unless causing a puff release of fission product, will only hasten the destruction.

A power excursion not caused by loss of cooling water is possible, but appears to be of almost negligibly small probability. Such an excursion will not become disastrous unless a significant fraction of the cooling water is boiled out of the reactor.

The scope of projects CG-553 and CG-600, a discussion of reactor hazards, a technical summary of pertinent aspects of reactor control and reactor cooling, and a discussion of development programs designed to increase efficiency of operation and further decrease the hazards are included in Volume 1.

VOLUME 2

Descriptive material supplying background information to support the technical summary in Volume 1 is included in Volume 2, as is a discussion of the release of fission products and damage to be expected under normal and extraordinary conditions. A bibliography for both volumes is included in Volume 2.

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1. PRELIMINARY

1.1 INTRODUCTION

This document is designed to accomplish three things: First, to summarize the current status of reactor safety at Hanford; second, to describe the expected status of reactor safety at Hanford after the reactor plant has been modified under Projects CG-558 and CG-600 and the power levels have been increased to the projected values; and third, to supply sufficient background information relative to the reactors and their surroundings so that the overall picture of the reactor safety problem at Hanford may be clearly defined.

In order to accomplish these objectives the first part of this report (Volume 1) includes a summary of the modifications to be accomplished under Projects CG-558 and CG-600, a discussion of the principal hazards associated with the Hanford reactors - both before and after modification, a summary of pertinent aspects of reactor control and reactor cooling, and a discussion of development programs designed to further decrease reactor hazards. This section of the report presumes a rather complete familiarity with Hanford reactors. The reader who is not acquainted with these reactors might well turn first to the Appendixes (Volume 2) which make up the second part of this report.

In the Appendixes is assembled the descriptive material which supports the technical discussions in Volume 1, and a discussion of the release of fission products and damage to be expected under ordinary and extraordinary conditions.

Throughout this report the emphasis is on the older reactors (B, C, D, DR, F, and H) although many of the discussions apply equally well to the K reactors. The K reactors, which are not affected by projects CG-558 and CG-600, are not specifically considered in this report.

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1.2 SUMMARY AND CONCLUSIONS

A disaster at Hanford would, by definition, involve the large scale release of radioactive fission products to the surrounding country-side. Such a release can occur only if the fuel elements (slugs) melt, and perhaps oxidize, in the reactor. Prevention of slug melting is thus synonymous with prevention of a disaster. Two possible ways in which slugs could be melted can be visualized:

1) If the reactor power were placed on a very fast rising period and allowed to rise without adding control, the heat generation rate would soon exceed the rate of heat removal by the cooling water and melting of the fuel elements would result. Thus one of the primary objectives at Hanford is to maintain strict control over the nuclear reaction at all times. Such control is accomplished by control mechanisms, sensory instrumentation, operating procedures and training so that the process is under constant manual control. In addition, independent, automatic control mechanisms are installed to provide automatic shutdown in the event of operator error, instrument failure, or unpredictable incidents such as loss of electrical power, or earthquake.

2) If the coolant supply to the reactor is permanently cut off, the fuel elements will melt. The time interval between loss of coolant and fuel melting may vary somewhat depending upon whether the reactor was operating or not when the water supply failed, but the end result will be the same -- release of fission products. Thus the second primary objective at Hanford is to provide an unfailing source of cooling water. This aim is attained by providing multiple water sources including a reserve supply of filtered water sufficient to provide 30 minutes of normal operation, a transition to shutdown conditions, and 24 hours of shutdown cooling.

The mechanisms, instruments and procedures to be used to maintain both control and adequate cooling will be considered in detail, as will the probable results of major failures.

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Projects CG-558 and CG-600 are modifications to the older Hanford reactors which will permit power level increases of as much as 50 per cent. These higher power levels will produce changes in the hazards associated with reactor operation. First, the fission product inventory in the reactors will be increased somewhat. It should be noted that the long lived fission product concentrations are primarily a function of the exposure of the uranium rather than of the power level. Since at the post project levels, the increased fission product inventory is due almost entirely to the short lived isotopes, the long term effects on the surrounding country side are scarcely changed by the increase in power level.

The increased power level will put more stringent requirements upon the speed with which control may be inserted. Higher power means a shorter time for water removal following loss of coolant flow at operating levels. Since removal of water from the reactor will in certain circumstances, result in large reactivity gains, the response of the safety system must be faster at higher powers if a power excursion is to be prevented.

If the coolant flow is lost after a shutdown, the higher power levels will simply accelerate the succession of events which follow. The melting times of various components will be reduced but no qualitative changes are expected. An exception to this statement might occur if the reactors were heavily enriched to provide more reactivity. In this case the total control capacity might become inadequate to hold the reactor subcritical under all circumstances.

Although the consequences of a disaster would be more severe under post project conditions, the probability of occurrence of a disaster will be reduced. The improved water plant and reactor piping system should reduce materially the chances of losing the cooling water. Increased safety in terms of control over the nuclear reaction will be achieved by instrument improvements. Chief among these are the sub-critical monitors which, for the first time in Hanford history, will provide a reliable measure of reactivity status before the reactors actually reach a critical condition during start up.

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In conclusion, the over-all hazards associated with operation of the Hanford reactors are expected to be decreased because of projects CG-558 and CG-600. Efforts are being made to define the variables in safety studies more closely in order to provide a more realistic picture of actual conditions and to point out the action which must be taken to provide increased safeguards against a disaster.

1.3 SCOPE OF PROJECTS CG-558 AND CG-600

The modifications to be made on the old reactors are summarized below. Project CG-558 refers to modifications to B, D, DR, F and H reactors while project CG-600 refers only to C reactor. It will be noted that the CG-558 modifications are much more extensive than those being made at C reactor which is already being provided with about 1.7 times as much cooling water as the other reactors. Thus the power level increases possible under Project CG-558 are large compared with the increase possible at C reactor.

	<u>MODIFICATIONS</u>						<u>REACTOR</u>					
	B	D	DR	F	H	C						
Modifications external to Reactor Building to increase flow of process water	x	x	x	x	x	x						
New front face piping (risers, crossheaders, etc.)	x	x	x	x	x							
New front face nozzles and pigtails (including venturies and/or orifices) for each tube	x	x	x	x	x							
Replace Downcomers	x	x										
Replace Panellits with improved models	x	x	x	x	x							
Provide thermal shield positive flow monitoring	x	x	x	x	x							
Provide comprehensive biological shield temperature monitoring	x											
Provide limited biological shield temperature monitoring		x	x	x	x							
Install poison column charge-discharge equipment which may be used during reactor operation	x	x	x	x	x	x						
Install new horizontal control rods	x	x	x	x	x							
Miscellaneous instrumentation changes to accommodate higher power levels (including providing or modifying an automatic power calculator)	x	x	x	x	x	x						

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The approximate operating capacities before initiation and ultimately after completion of Projects CG-558 and CG-600 are shown below. The post project power levels include increases contingent upon relaxation of existing limits. This relaxation is expected to result from technical development programs now being pursued.

REACTOR	BEFORE PROJECTS			AFTER PROJECTS		
	Number of Effective Tubes	Flow Rate (GPM)	Power Level (Megawatts)	Number of Effective Tubes	Flow Rate (GPM)	Power Level (Megawatts)
B	1350	48,000	900	1400	74,000	1800 max.
C	1500	83,000	1700	1500	88,000*	2000 max.
D	1350	53,000	1000	1400	74,000	1800 max.
DR	1350	50,000	1000	1400	74,000	1800 max.
F	1350	47,000	1000	1400	74,000	1800 max.
H	1450	57,000	1200	1500	74,000	1900 max.
* By modifying the front face piping, pigtails and nozzles at C reactor under a subsequent project the flow rate can be increased to 94,500 gpm.						

Interpreting this table, it is seen that the operating conditions at B, D, DR, F and H reactors after completion of Project CG-558 will be quite similar to current operating conditions at C reactor. Furthermore, the indicated average power increase of eighty per cent for the B, D, DR, F and H reactors will result from (a) a four per cent increase in the number of effective central tubes, (b) an average increase in coolant flow of forty-six per cent, and (c) a potential thirty per cent average increase in power level due to relaxation of existing limits. At C reactor completion

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of Project CG-600 will result in (a) no increase in the number of effective central tubes, (b) a six per cent increase in coolant flow, and (c) a potential eleven per cent increase in power level due to relaxation of existing limits.

The power levels thus obtained were used in evaluating the hazards to be encountered following completion of Projects CG-558 and CG-600.

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- VOL. 3 Design Criteria 100-H, Area.
- VOL. 4 Design Criteria 100-F Area.
- VOL. 5 Codes, Standards and Bibliography.

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2. REACTOR HAZARDS

2.1 GENERAL CONSIDERATIONS

It is believed that no disastrous release of airborne fission products is possible as long as adequate cooling water can be supplied to a Hanford reactor. Conversely, loss of cooling water (unless the reactor has been shut down for a very long time prior to the water loss) will almost surely result in destruction of the reactor and probably in the large scale release of fission products. This destruction will occur even though the reactor is made subcritical by insertion of the safety rods immediately upon loss of water. Consequently, the discussion of reactor hazards will be concerned primarily with events following water loss under various conditions and possible mechanisms leading to water loss. The elaborate precautions taken to assure continuity of water flow will be discussed in the section on reactor cooling.

It is believed that operation of the reactors will be safer after completion of projects CG-558 and CG-600 than it is at present. The two principal reasons for this conclusion are: A larger safety factor in the strength of the front risers combined with check valves in the front-face crossheaders will reduce the probability of a disastrous water loss. In addition, the use of new and improved instrumentation, especially the proposed "sub-critical monitor", will reduce the probability of a power excursion.

In the unlikely event of loss of control, or loss of cooling water, however, the time available for remedial action will be reduced at the higher power levels to be attained following completion of these projects. The fission product inventory and the effects of a disaster will be greater.

Limited release of fission products would result from inadequate water flow in one, or a few, process tubes; or from burning of a few fuel elements which were improperly cooled after being discharged from a reactor. Contaminated effluent water would be diverted to filter beds,

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or "cribs", whence a greatly reduced quantity of fission products would eventually reach the river at a very slow rate. The airborne contamination would probably affect only the immediate vicinity of the reactor. Such limited release of fission products by these, or similar, events would be classed as operational "incidents" and will not be considered further in this section.

2.2 EVENTS FOLLOWING LOSS OF PROCESS COOLING WATER

2.2.1 Introduction

The consideration of events following loss of cooling water at a Hanford reactor is highly speculative. The discussion which follows is a synthesis of "educated guesses" founded upon occasional fragments of experimental data. It is anticipated that the prediction uncertainties for cases in which water pressure is lost while at operating levels will be greatly reduced when results are obtained from the analogue computer currently being purchased. This computer will be supported by experimental programs as discussed in the "heat transfer" part of the section (5.5) on development programs.

Because of the uncertainties involved in predicting the behavior of a Hanford reactor under violent transient, or "disaster", conditions it has been customary to assume the most dangerous credible conditions. Consequently, the tentative conclusions reached in this discussion are probably pessimistic.

2.2.2 Power Excursion

Immediately following loss of water pressure the water in the central region of the highest power process tube would begin to boil, expelling liquid water from the tube. As the water is expelled, the reactivity of the reactor would increase and the power generation rate would begin rising exponentially. As the power level rises the rate of conversion of water to steam would increase, and so on. The process is self accelerating.

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The magnitude of a power excursion following loss of water pressure depends critically upon (1) the rate at which water would boil out of the process tube, (2) the increase of reactivity upon water loss, and (3) the speed with which the vertical safety rods shut down the reactor.

It has been estimated that total and instantaneous loss of pressure would cause the water to boil out of a process tube generating heat at the rate of 300 KW in 2.5 seconds; and out of a 1300 KW tube in 0.7 seconds. The latter number approximates post CG-558, CG-600 conditions. Preliminary experimental results are consistent with these estimates.

For the range of exposures contemplated for post project operation the reactivity jump upon loss of water decreases as the temperature of the graphite increases, and, for a given moderator temperature increases as the exposure of the uranium increases. Exposure of uranium is not expected to affect significantly the reactivity jump upon loss of water in a "cold" reactor. The exact values are uncertain, but approximate values are given in Table III of the section (3) on reactivity control. For B, D, DR, F and H reactors loss of water during startup (when the graphite is cold) would cause a reactivity jump of about 750 inhours for all contemplated exposures. At operating temperatures the reactivity jump on loss of water would be about 75 inhours for 100 MWD/T weighted average exposure, increasing to about 300 inhours for 500 MWD/T average exposure. At C reactor each of these values is increased by about 150 inhours. Since a reactivity jump of about 280 inhours will make the reactor "prompt critical", extremely short rising periods are possible. It should be emphasized that considerable uncertainty still exists in these numbers because of continuing uncertainty in the magnitude of the effect of exposure on the moderator temperature coefficients of reactivity.

The vertical safety rods have ample strength to compensate for the reactivity jump following loss of water and to shutdown the reactor rapidly. Excessive fringe enrichment in B, D, DR, and F could create an exception under special conditions. The amount of fringe enrichment in these reactors

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is limited however and this exceptional case is thus eliminated. The vertical safety rod strength is shown in Table III of the section on reactor control and the speed of action of the safety rods is shown in Table IV of the same section.

Experiments on scram transients ⁽¹⁾ showed that about two seconds elapsed between the time the button was pushed to actuate a scram and the time at which the recorder showed a significant flux decrease. The two second figure is subject to error because of human reaction time and lag in response of the recorder. Once the flux starts to decay, the decrease is very rapid.

Combining the above facts leads to the conclusion that under special circumstances a severe power excursion could occur. The water pressure to the entire pile would have to be lost instantaneously and the reactivity jump following water boil out would have to be fairly large - 300 ih or higher. In addition, the power level would have to be high enough to provide rapid boilout of the water in the process tubes. Under these circumstances a fast rising period could be produced which would last long enough to produce melting of fuel jackets during the excursion.

It must be understood that the above conditions must all be satisfied however. For example, loss of water pressure by a linear decay over a period of as little as 5 seconds would surely provide enough extra time before the water boils out of the process tubes to permit entry of the VSR's with no power excursion even at the highest contemplated post Project CG-558 tube powers. Even shorter pressure decays would be highly beneficial in this respect although detailed calculations will be necessary to provide accurate numbers. In addition it should be noted that high tube powers and large reactivity jumps upon loss of water do not normally occur together. For projected Hanford exposures the maximum reactivity jump will occur with cold graphite. The normal rate of power rise is slow enough that the graphite temperature increases with power. Hence as the possible rate of water boil-out increases, the consequence of boil-out in terms of reactivity gain decreases. ⁽⁵⁾

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In summary, the following conclusions appear valid:

(1) Under very special combinations of assumed circumstances a severe power excursion is possible and the higher power levels of post 558 operation will increase the size of that excursion.

(2) Under more reasonable circumstances, a power excursion sufficient to melt fuel jackets is not probable. Post CG-558 operation is not expected to change this conclusion.

2.2.3 Fission Product Heating

Even if no power surge occurs during the shutdown transient or if the reactor is shutdown before process water pressure is lost, the reactor would still be destroyed by fission product heating. Since the occurrence of a serious power excursion would almost certainly be initiated by irreparable damage to process water piping, re-establishment of water flow would be impossible and the destruction of the reactor would be inevitable. Thus the only effect of a power excursion (assuming that it was not violent enough to cause a puff discharge of fission products) would be to compress the time scale given below.

The time required for the residual fission product heating to melt the various pile components has been predicted by Ekern⁽²⁾ as follows:

TABLE I

PREDICTED MELTING SCHEDULE

(Time of melting measured from instant of water pressure loss)

Event	Tube Power (KW/Tube)	Time Shutdown Before Water Loss			
		Zero	30 sec.	30 min.	10 hr.
Jackets Melt	800	20 sec.	6 min.	15 min.	9 hr.
	1300	4 sec.	3 min.	8 min.	3 hr.
Tubes Melt	800	2 min.	11 min.	2 hr.	12 hr.
	1300	30 sec.	4 min.	20 min.	7 hr.
Uranium Melts	800	6 hr.	6 hr.	9 hr.	21 hr.
	1300	2 hr.	2 hr.	7 hr.	13 hr.
Vertical Safety Rods Melt	800	7 hr.			
	1300	3 hr.			

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It should be emphasized that these numbers are not precise. A time of three hours should be interpreted as meaning 'a few hours'. However, two conclusions may be drawn from this table. As the power level is increased the time available for remedial action is decreased. And, of greater significance, as the interval between reactor shutdown and water loss is increased the time available for remedial action is increased.

Since the rate of fission product heat generation decays exponentially the first few seconds of shutdown time have the greatest effect. Thirty seconds of downtime before water loss extends the additional time available for remedial action before extensive damage from seconds to minutes; hours of downtime extends this time only to hours.

2.2.4 Reactivity Changes

It is not certain that the shutdown reactor would remain sub-critical indefinitely following loss of water. Figure 1 shows the expected reactivity variation following water loss. This graph is based on a central-tube power generation rate of 800 KW, prior to simultaneous water loss and shutdown. It is further assumed that no initial power excursion occurs. The effect of increasing the power level to 1300 KW per central-tube would be principally to change the time scale as indicated in Table I. The most striking feature of the graph is the effect of exposure.

For zero exposure both the dry graphite temperature coefficient and the metal coefficient are negative. Thus as the temperature rises the reactivity of the reactor falls until the uranium melts. At this time the reactivity drops still further because of the change in the shape of the uranium, rises as the xenon escapes, and then continues to fall as the temperature increases. Therefore the reactor loaded with low-exposure fuel is expected to remain sub-critical at least as long as the vertical safety rods remain effective.

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FIGURE 1
Representative Reactivity Status of Reactor
Following Simultaneous Shutdown and Loss of Water
From Full Power
(800 KW/Tube; 1300 Effective Tubes)

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As the exposure increases, removing water and inserting safety rods will leave the reactor with a smaller negative reactivity than for zero exposure. The most striking effect, however, is the expected change in the dry graphite coefficient from negative to positive. Once the process tubes melt and good thermal contact with the graphite is attained, the effect of fission product heating is to increase the reactivity towards critical. By the time the xenon has escaped or decayed the reactor will be much closer to critical than it would have been with low exposure uranium. Beyond this point the reactivity may continue to increase as the temperature rises. However, the melting of the uranium slugs may introduce large negative reactivity effects.

A complete analysis of the events following uranium melting is not possible at this time. However, the following ideas appear plausible. During the interval between the time the water loss occurs and the time the uranium melts (6 to 10 hours), the backlog of iodine will be decaying into xenon and the poisoning effect of the xenon will be increased. (Note that this effect has not been included in the previous chart.) Upon reaching the uranium melting point, the xenon and iodine will presumably escape from the slugs into the graphite stack. In this dispersed condition, the xenon is and even more effective poison than while held in the slug. Some of the xenon could diffuse into the graphite prior to slug melting - the amount to be expected is still uncertain.

The effect of uranium melting upon reactivity is not precisely known. If the slugs simply "slump" in the process channels the reactivity loss is relatively mild. On the other hand, if the molten uranium flows into cracks and holes in the graphite as did the melted portions of the tube at K reactor⁽⁴⁾, then the reactivity losses will become appreciable and could easily become so large that the presence or absence of the control columns would be immaterial.

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The effect of reaching a critical configuration during the heating and melting interval is of interest. The effects which could produce a critical condition are the positive graphite temperature coefficient (for high exposure) and the escape of xenon from the graphite stack. Both of these effects would be relatively slow acting and no very short rising period would be observed. The effect of the heating would then be to accelerate this melting of the uranium and its dispersal through the graphite which is expected to produce a large negative reactivity effect. Thus even if the pile does regain a critical configuration, no large increase in the fission product release is to be expected. The fission products generated during the critical period would be very small compared to the fission product inventory already present in the high exposure uranium. It should be mentioned that the above discussion would not apply to a J-N or J-Q load.

The effect of the projected power level increases will be to accelerate the succession of events following water loss but will not introduce and qualitative changes in those events. Thus the principal effect will be to reduce the time available for any given remedial action which might be contemplated.

2.2.5 Chemical Reactions

Both graphite and uranium will oxidize at the temperatures found inside an operating reactor. Thus if air leaks into the reactor following expulsion of the process cooling water the oxidation reaction may contribute both to the heating of the reactor and the release of fission products. These effects will probably be small, however, as long as the reactor remains enclosed and the reactor atmosphere stagnant.

Energy is stored in low temperature graphite during irradiation. This energy is released when the temperature of the graphite is raised to an "annealing" temperature. Since the current and post project graphite temperatures are above the annealing temperature, except possibly in the peripheral layers of graphite, the amount of heat added to the shutdown reactor from the energy stored in the graphite is expected to be small, and perhaps negligible.

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Aluminum and uranium inter-diffuse at the melting temperature of aluminum, forming a high melting-point alloy. Since this alloy has poor heat conducting qualities as well as a high melting point, it will probably result in increased containment of fission products up to the melting temperature of the uranium, which would be reached in a shorter time. This effect is not expected to be large, but it has been included in Ekern's calculations.

This aluminum-uranium alloy is significant also in that it oxidizes in contact with steam, forming a tight plug in the process tube. Such a plug would effectively prevent restoration of cooling water to the damaged process tube, but it might also permit restoration of cooling water to undamaged tubes without simultaneously bringing large amounts of water into contact with the molten uranium and the hot graphite. The oxide plug would also inhibit the escape of fission products from the reactor.

2.2.6 Fission Product Release

The quantity of radioactive fission products which would be released from a Hanford reactor is highly speculative. If the reactor shield and the reactor building remain tight, and if the reactor and reactor building atmospheres remain stagnant the release will be minimized. Early experiments suggest that fission products are very strongly absorbed by or adsorbed on graphite, even at elevated temperatures. If the process tubes are effectively sealed by aluminum-uranium oxide plugs the fission products would have to seep through the graphite and up the stack. Under these conditions the total release might be less than one per cent of the reactor inventory, and the rate of release very slow. On the other hand, if the active core of the reactor were exposed to the air so that graphite and uranium oxidation would produce particulate matter to be carried aloft by convection currents, the release might reach fifty per cent of the fission product inventory, and occur at a relatively rapid rate.

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2.3 POWER EXCURSION NOT CAUSED BY WATER LOSS

A power excursion would not become disastrous unless the power surge was so violent that process water was lost from a significant number of process tubes. Control equipment is designed and operated in such a manner that rapid power increases are avoided. Accidental discharge of one process tube full of poison, for example, would lead to a power excursion with a sufficiently long rising period that the excursion would be quickly terminated by the automatic insertion of the vertical safety rods. Extraordinary, and improbable, events would be required to initiate a disastrous power excursion which was not initiated by loss of process cooling water. Such an excursion would result in process water boilout from the central region of the reactor, and the sequence of events in this region would be similar to those discussed for loss of water.

The effects of a power excursion without complete loss of water would be, in general, less drastic than if process water had been lost. The tube damage would be limited to a smaller area, the graphite temperatures would be limited to smaller values, and the reactor probably would not become supercritical after being shutdown.

2.4 EARTHQUAKE

2.4.1 Earthquake Potential at HAPO Site

All of the State of Washington east of the Cascade Mountains is regarded as an area of moderate earthquake potential. This implies the possibility of earth tremors of intensity MM7 or MM8 on the Modified Mercalli Scale of 12. On this scale MM6 is accepted as the threshold of minor structural damage. The high rating is largely based on the fact that much of the building is done on unconsolidated sediments which produce more violent earthquake motions at certain frequencies that would be experienced on solid bedrock.

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Sharp fluctuations of the water levels in wells in the Hanford area have been correlated with recorded earthquakes with epicenters as distant as India and Israel. ⁽³⁾ Quakes near Ketchikan, Alaska, and in Seattle in 1949 apparently registered about MM6 at Hanford. Of fourteen tremors recorded at Hanford between 1949 and 1955, four are believed to indicate shocks of local origin since they were not detected elsewhere. Some of the tremors appeared to approach MM6 in intensity. An earthquake of intensity MM6 was reported at Corfu, Washington, fifteen miles north of Hanford townsite in 1918.

2.4.2 Events Following an Earthquake

Each reactor is supplied with seismoscopes connected into the safety circuit of the reactor. The seismoscopes are set to trip the circuit for a tremor of intensity MM5 or greater. Coincident signals from two seismoscopes are required to shutdown the reactor. At B, D, DR, F and H reactors the seismoscopes trip the safety rods into the reactor. At C reactor the seismoscopes trip the Ball 3X hoppers.

If the earthquake does not interrupt the process water flow, no power excursion will result. If the earthquake does interrupt the water flow, the front riser water pressure decay will trip the Ball 3X hoppers at B, D, DR, F, and H reactors; and the panellit tube-pressure gauges will trip the vertical safety rods at C reactor. Subsequent events will follow the pattern previously discussed under "loss of water".

2.5 FLOOD

Normal Columbia River floods are not expected to initiate any disaster conditions. The floor of each river pump house is about fifteen feet above the 1948 flood level. The reactor control room floor is still higher, varying from approximately twenty-five to sixty-feet above the 1948 flood level. The high pressure pumphouse and the filtered water storage basins are also well above the level of the river pumphouse.

A high flood, such as would be caused by failure or destruction of the Grand Coulee Dam would result in loss of cooling water. Perhaps

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twelve hours warning would precede water loss. Supplementary cooling systems have been considered, but, as discussed in the section on development programs, have not been advanced beyond the cost scoping stage.

2.6 MISCELLANEOUS HAZARDOUS CONDITIONS

The probability of occurrence of potentially hazardous incidents such as fire, storm, strike, or riot is not affected by projects CG-558 and CG-600. Nor is the probability of occurrence of electric power failure, operator mistake, or act of war. The probability of equipment or instrument failure or malfunction will probably be decreased, due to installation of new and improved instrumentation, and the unusual opportunity for preventative maintenance occurring during the reactor shutdown for completion of these projects.

Of these potentially hazardous incidents, only an act of war is likely to result in a reactor disaster. Destruction of the Grand Coulee Dam would be, and direct bombing of the area could be, disastrous.

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3. REACTOR CONTROL

3.1 INTRODUCTION

As noted in the previous section, one of the two major problems at Hanford is the maintainance of adequate control over the nuclear reaction. In the following section, this problem will be discussed in considerable detail. First of all, the "normal" requirements which the control system must meet will be presented in a discussion of normal operational reactivity transients. Included here are descriptions of the various reactivity effects, the shutdown transients observed and the various methods of returning a Hanford reactor to full power following outages of various lengths.

In the second part of this section, the control and safety systems are described and their capacities and response times given.

The instrumentation used in the Hanford reactors is described in Appendix III.

3.2 NORMAL OPERATIONAL TRANSIENTS

3.2.1 Sources of the Transients

The reactivity of a Hanford reactor depends upon (1) the lattice design, (2) the particular loading of natural uranium, enriched uranium and poison in the reactor, (3) the position of the control rods, (4) the temperature of the metal and of the graphite, (5) the quantity of xenon in the reactor, (6) the presence of water in the reactor, and (7) the composition of the reactor atmosphere. U-235 burnout, and Sm-149 and Pu-239 buildup within the reactor give rise to "long term reactivity effects", which occur at such slow rates that they are not usually regarded as operational transients. The quantity of xenon in the reactor, however, changes with comparative rapidity during "non-equilibrium" operation and gives rise to severe reactivity transients. Significant, but generally smaller transients are associated with temperature changes in fuel and moderator.

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The effect on reactivity of changing the temperature of the graphite moderator depends upon the exposure of the uranium charge. For zero exposure, the "dry" Hanford reactor has a negative moderator coefficient of reactivity while the addition of water to the cooling annulus results in a positive moderator coefficient. The effect of buildup of Pu-239 in the uranium is expected to change both the wet and the dry graphite coefficients in the positive direction. The change in the "dry" coefficient is believed to be larger than the change in the "wet" coefficient. Thus, although the green Hanford reactor, i. e., a reactor loaded with zero exposure uranium, should gain a smaller amount of reactivity upon water loss while the graphite is hot than while it is cold, this is not necessarily true at high exposures. Indeed, there is believed to be an exposure at which the "dry" coefficient is actually positive and larger than the "wet" coefficient. For such an exposure, the reactivity gain upon loss of water will increase steadily with graphite temperature. Contemplated Hanford exposures are believed to be below this critical exposure.

Temperature changes in the uranium metal also affect the reactivity of the reactor. Some neutrons will be captured in uranium-238 while slowing down through the resonance capture region. As the temperature of the uranium rises, the greater random motion of atoms increases the effective width of the uranium-238 capture resonance region. This results in increased capture of neutrons and fewer neutrons slowing down to take part in the fission process. Consequently, an increase in fuel temperature is accompanied by a decrease in reactivity.

The magnitudes of these temperature induced reactivity changes are listed in Table III, (page 41).

3. 2. 2 Shutdown Reactivity Transients

After a reactor has been operating at a given power level for some time under equilibrium conditions and the horizontal control rods are inserted slightly into the reactor, making it subcritical, the neutron flux

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level decreases. However, as the fission rate declines the uranium metal cools, causing an increase in reactivity which must be compensated by further rod movement. The cooling of the graphite causes a decrease in reactivity in the water cooled reactor, independent of rod movement, which drives the reactor further subcritical. Finally, the quantity of xenon in the reactor increases rapidly for two reasons: Iodine-135 present in the reactor continues to decay into Xenon-135, and the rate of removal of

Xenon-135 is determined by the rate of decay of the xenon only, the neutron capture process having effectively ceased. The buildup of Xenon rapidly drives the reactor further subcritical until after about three quarters of an hour even the complete withdrawal of the control rods cannot restore the shutdown reactor to the critical condition. Eventually the iodine backlog decays into xenon and the xenon decay proceeds until the reactor can again be made critical by complete rod withdrawal. Typical shutdown reactivity transients are shown in Figure 2.

The minimum downtime depends upon both the reactor power level at equilibrium and the exposure of the uranium. The residual reactivity (R_{∞}), which is defined as the reactivity remaining after graphite cooling and complete xenon decay, depends primarily upon exposure. Representative values for minimum downtime and residual reactivity for limiting values of power level and uranium exposure are shown in Table II.

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TABLE II
REPRESENTATIVE VALUES FOR MINIMUM DOWNTIME
AND RESIDUAL REACTIVITY ***

Before and After Projects CG-558 and CG-600

Average Exposure * →	Minimum Downtime		Residual Reactivity (R_{∞}) **	
	100 MWD/T	500 MWD/T	100 MWD/T	500 MWD/T
	(hr)	(hr)	(ih)	(ih)
Before Projects (Relative Power = 1.0)	22	33	1050	500
After Projects (Relative Power = 1.8)	29	42	1050	500
<p>* All exposures are $(\text{flux})^2$ weighted cumulative exposures averaged over the entire fuel loading.</p> <p>** R_{∞} is defined as the reactivity remaining after graphite cooling and complete xenon decay.</p> <p>*** These values are based upon the assumptions that (a) 400 inhours worth of poison is held in the control rods at equilibrium, and (b) no reactivity changes are made during shutdown.</p>				

The minimum downtimes listed in Table II could be reduced by increasing the control rod excess held during operation, discharging exposed uranium with its iodine and xenon, or discharging poison columns, i. e., process tubes loaded with neutron absorbing materials. Poison column control facilities are authorized under these projects to supplement the horizontal control rod systems. The poisons used are customarily chosen so that neutron capture produces valuable isotopes. In the event that poisons may economically be discharged during shutdown and new poisons recharged after the reactor has attained criticality but before equilibrium power levels are reached, the minimum downtime will approach thirty hours for high exposure fuel loadings.

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FIGURE 2

Representative Shutdown Reactivity Transients
B, C, D, DR, F and H Reactors
(For natural uranium loading,
or minimum loading enrichment)

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Variations in the amount by which the reactor is driven subcritical following shutdown before and after Projects CG-558 and CG-600 are not of great importance. In neither case does it presently appear feasible to provide sufficient reactivity to over-ride the xenon poisoning so that the startups could be made at any time after shutdown.

3. 2. 3 Startup Reactivity Transients

(1) Xenon Free Startup

After the reactor has been shutdown for about three days, the xenon poison has essentially achieved complete decay. To bring the reactor up to power under this condition the safety and control rods are slowly withdrawn, in sequence, until the reactor becomes "critical". The fuel immediately rises in temperature, causing a decrease in reactivity which is compensated by further rod withdrawal. As the graphite heats up the reactor increases in reactivity. This increase must be compensated by control rod insertion. Xenon poison buildup is delayed, as the parent fission product, Iodine-135, decays with a six-hour half life. When xenon is formed the reactivity decreases and control rods are gradually removed until xenon equilibrium is reached.

The magnitude of the reactivity transients involved in a startup depend primarily upon the power levels attained. Potentially, the net increase in reactivity is greater than the horizontal control rod strength available. Therefore selected process tubes are temporarily charged with neutron absorbing material to reduce total reactor reactivity during the startup transients and the power level is increased by steps, each step being held until xenon buildup provides additional control.

Representative startup reactivity transients are shown in Figure 3. The transient curve labeled "After Projects CG-558, CG-600" corresponds to low exposure (100 MWD/T (flux)² weighted exposure averaged over the fuel load) and represents the most severe transients expected. For higher average exposure (500 MWD/T) the transients will roughly correspond to

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FIGURE 3

Representative Startup Reactivity Transient
B, C, D, DR, F and H Reactors
(reactor initially free of xenon)

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the curve labeled "Before Projects CG-558, CG-600". The actual average exposure will probably be somewhere between these two limiting valves.

It is seen that the startup transients will increase in severity following completion of Projects CG-558 and CG-600. This will increase the hazards of operation only insofar as the increased use of supplementary control (poison columns) will increase the possibility of accidental discharge of poison from the reactor.

(2) "Cold" Startup

After the "minimum downtime" has elapsed, the decay of the xenon burden in the reactor permits the reactor to again be made critical. By this time the graphite and metal have cooled. Part of the xenon poison, however, remains in the reactor. A startup under these conditions is called a "cold" startup, and is of routine occurrence.

Compared to the xenon-free startup, the cold startup is characterized by (1) lower initial reactivity, compensated by rods, (2) steadily increasing reactivity, (3) a more rapidly rising reactivity transient after criticality is reached, due to xenon burnout, and (4) a larger uncertainty in the initial value of the reactivity. The greater uncertainty in the initial value of the reactivity results from inaccuracy in predicting the xenon transients, compounded by loading changes during shutdown.

The reactivity transient after criticality is reached follows the general pattern of the transient for the xenon-free startup. The rising transient is steeper for the cold startup, however, due to the xenon burnout. The potential maximum value of the reactivity is decreased somewhat due to xenon remaining in the reactor. Poison is charged to supplement the control rods, and stepwise power increases are used, as for the xenon free startup.

(3) "Hot" Startup

Referring to the curve of shutdown reactivity transients, Figure 2, it is seen that for a relatively short time after shutdown - perhaps three-quarters of an hour - sufficient reactivity remains in the reactor to permit

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bringing the reactor to critical. A startup during this period is called a "hot" startup.

At the time of initiation of the startup procedures, the uranium is cold, the graphite is cooling, the xenon content of the reactor is increasing, and the reactivity of the reactor is decreasing very rapidly. Prompt and decisive action must be taken to bring the reactor to critical. A limited amount of poison may be discharged in order to extend the time available for startup.

After the reactor is made critical the reactivity transients follow the general pattern discussed for the other classes of startups. The rising transient is steeper due to the additional xenon available for burnout. Stepwise increases in power level and loading changes are used as before to control the increase in reactivity of the reactor during the approach to equilibrium conditions.

3.3 CONTROL AND SAFETY ROD CAPACITY

The effectiveness of a control or safety rod depends upon its geometry, its composition, and the relative value of the neutron flux into which it is inserted. Thus a given rod inserted into a high flux region of a reactor is more effective in reducing the reactivity than a similar rod inserted into a low flux region. The flux distribution in a reactor is modified by the location of enriched materials, and of poisons, in the reactor. Therefore, a discussion of control and safety rod capacity must be prefaced by consideration of loading patterns.

In a Hanford reactor loaded only with natural uranium, the neutron flux is a maximum in the center, falling off towards the front, rear, top, bottom and sides approximately as the cosine function. Charging poison into the central region depresses this peak, allowing the average flux to be increased without exceeding the central tube limits. In addition, charging U-235 enriched material in the peripheral tubes increases the relative

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neutron flux near the top, bottom, and sides of the reactor. This permits a further increase in the average flux without exceeding the central tube limits. The allowable flux is limited by the rate at which heat can be removed from a process tube by the cooling water.

The use of fringe enrichment increases the reactivity of the reactor, and with compensating poison, reduces the relative flux near the center of the reactor. As a consequence of the flux redistribution resulting from fringe enrichment the vertical safety rods are less effective in holding the reactor subcritical when enrichment is used.

The vertical safety rod control capacities calculated for maximum and minimum allowable enrichment following completion of Projects CG-558 and CG-600 are shown in Table III. These values have been calculated from consideration of the material buckling required to make the dry-reactor critical with the safety rods in the reactor and all other controls out. They represent, then, ultimate "holding power" of the safety rods in preventing a disaster as the reactor approaches critical with all the safety rods in. "C-Columns", the present enrichment loading, are process tubes charged with U-235-aluminum alloy slugs containing approximately four and one-eighth per cent U-235. The effective safety rod control capacity before completion of Projects CG-558 and CG-600 is approximately as shown in the column headed "500 MWD/T". (It is expected that "C" column loadings will be replaced during the next few months by lightly enriched uranium "E" metal; control capacity effects should not be changed appreciably by this substitution, but the use of "E" metal has safety advantages over "C" metal in event of melting following water loss.)

Also shown in Table III are the enrichments corresponding to the two average exposures tabulated, and the maximum values of reactivity change resulting from the various transient effects as the condition of the reactor is changed. The initial condition of the reactor in each case is equilibrium operation at post project power level, with normal water flow in the process tubes, i. e., "hot", "wet" condition. The first column, labeled "cold, wet"

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TABLE III

MAXIMUM REACTIVITY TRANSIENTS VS. SAFETY ROD CAPACITIES
AFTER COMPLETION OF PROJECTS CG-558 AND CG-600

(Transients are expressed in reactivity increase from the hot, wet reactor in equilibrium)

Reactor	B, D, DR, F						H						C					
	100 MWD/T			500 MWD/T			100 MWD/T			500 MWD/T			100 MWD/T			500 MWD/T		
	Cold, Wet	Cold, Dry	Hot, Dry	Cold, Wet	Cold, Dry	Hot, Dry	Cold, Wet	Cold, Dry	Hot, Dry	Cold, Wet	Cold, Dry	Hot, Dry	Cold, Wet	Cold, Dry	Hot, Dry	Cold, Wet	Cold, Dry	Hot, Dry
Enrichment (No. of C-Co umns)	50 - 100			0			120 - 140			~30			120 - 140			~30		
Condition	Cold, Wet	Cold, Dry	Hot, Dry	Cold, Wet	Cold, Dry	Hot, Dry	Cold, Wet	Cold, Dry	Hot, Dry	Cold, Wet	Cold, Dry	Hot, Dry	Cold, Wet	Cold, Dry	Hot, Dry	Cold, Wet	Cold, Dry	Hot, Dry
Xenon Loss (ih)	+1000	+1000	0	+1000	+1000	0	+1000	+1000	0	+1000	+1000	0	+1000	+1000	0	+1000	+1000	0
Graphite Cooling (ih)	- 675	- 675	0	-1250	-1250	0	- 675	- 675	0	-1250	-1250	0	- 675	- 675	0	-1250	-1250	0
Uranium Cooling (ih)	+ 350	+ 350	0	+ 350	+ 350	0	+ 350	+ 350	0	+ 350	+ 350	0	+ 350	+ 350	0	+ 350	+ 350	0
Control Rod Shadowing (ih)	+ 350	+ 350	+350	+ 350	+ 350	+350	+ 300	+ 300	+300	+ 300	+ 300	+300	+ 300	+ 300	+300	+ 300	+ 300	+300
Water Loss: (ih)	---	+ 750	+ 75	---	+ 750	+300	---	+ 750	+ 75	---	+ 750	+300	---	+ 900	+225	---	+ 900	+450
Net Reactivity Increase (ih)	1025	1775	425	450	1200	650	975	1725	375	400	1100	600	975	1875	525	400	1300	750
Safety Rod Capacity (ih)	1200	1200	1200	1800	1800	1800	1600	1600	1600	2800	2800	2800	2000	2000	2000	2800	2800	2800
Temporary Poison Required (ih)	---	575	---	---	---	---	---	125	---	---	---	---	---	---	---	---	---	---

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indicates the reactivity gained during a normal shutdown after the graphite has cooled and the xenon has completely decayed away. It should be noted that complete xenon decay requires about three days time. The second column labeled "cold, dry" indicates the reactivity to be gained should a normal shutdown be followed by loss of process cooling water. The third column, labeled "hot, dry" shows the immediate reactivity gain upon loss of process water in a reactor operating in equilibrium at design power.

It is expected that average exposures following completion of projects CG-558 and CG-600 will lie between 100 and 500 MWD/T.

It is seen from Table III that the vertical safety rods are capable of holding the wet reactor subcritical even after the graphite has cooled and the xenon has completely decayed away and that supplementary poison may have to be charged into the reactor to assure that the reactor would remain subcritical in the event of water loss. (Operating specifications require that sufficient temporary poison be charged concurrent with the shutdown transient so that the vertical safety rods may be withdrawn without making the reactor critical. In effect this means that the total safety rod capacity is reserved for a possible water loss emergency.) Further, it is seen that the safety rods have ample strength to shut down the reactor in the event of water loss at operating power levels. The catastrophic nature of events which would follow a complete water loss were discussed in Section 2.

The speed with which the vertical safety rods may be introduced into the reactor and the resultant rate of neutron flux decay (power level decay rate) must be considered in the evaluation of reactivity limitations. A limited amount of experimental data is available.^(1, 2) However, the values in Table IV must be considered as representative, rather than exact. The reactivity effects of the partially inserted rods were calculated by semi-empirical methods. The distance fallen and the relative flux were determined experimentally.

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TABLE IV
SPEED OF CONTROL - VERTICAL SAFETY RODS

Time After Manual Scram ⁽¹⁾ (Sec)	Time From Clutch Release ⁽²⁾ (Sec)	Distance Fallen ⁽³⁾ (Ft)	Reactivity Effect ⁽²⁾ (% of Total)	Relative Flux ⁽¹⁾ (ϕ/ϕ_0)
0.1	0.0	0.0	0.0	1.0
1.1	1.0	8.1	3	1.0
1.5	1.4	15.3	27	1.0
2.1	2.0	26.3	85	.8
2.5	2.4	29.1	95	.4
2.8	2.7	29.8	96	.2

It is seen from the table that during the first 1.5 seconds after the safety circuit is tripped the reaction is essentially unaffected. During the next second, however, the safety rods become about 95 per cent effective and the neutron flux is reduced to about half of its original value.

Recent control rod calibrations^(3,4) have disclosed that the horizontal control rods have about fifty per cent greater control value than previously believed to be true. These calibrations were based upon measurement of reactor periods at low power levels as a function of rod position. Simultaneous determination of the reactivity value of xenon poisoning disclosed that the xenon poisoning effect is about ten per cent greater - over the measured range - than had been used previously.

The control rod values, and the xenon poisoning values used in this report are based upon these recent calibrations.

The horizontal control rods are normally used with all the vertical safety rods withdrawn. Under these conditions, the total control capacity of the nine-element control rod systems at B, D, DR, and F reactors is

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about 600 inhours. At C and H reactors the 15 rod systems have a capacity of about 1000 inhours. The ultimate holding power of the horizontal control rod system in preventing a disaster as the reactor approaches critical with the safety rods in will be much less than these values, since the control rods will lie largely within the region of greatly depressed flux occupied by the safety rods and will thus be "shadowed" to a marked degree.

Knowledge of the control capacity of the partially inserted horizontal rods is of primary importance in establishing the initial reactivity balance during the approach to critical. The control rod strength is strongly affected by the positions of neighboring control rods, and to a lesser extent by variations in the reactor loading pattern. Control rod calibrations have been made for representative rod configurations and reactor loadings. However, due to inherent uncertainties in the rod calibrations and to loading variations, strict adherence to safe operating procedures is required in order that undesirable short initial reactor periods may be avoided.

The speed of the horizontal rod system is adequate to maintain control of the reactor after initial control has been established. Under post-project conditions, the control rods will be able to decrease the reactivity of any reactor at a greater rate than that at which reactivity will be gained during the most severe transients associated with xenon burnout and temperature effects.

3.4 PHILOSOPHY OF SPECIFYING THE APPROACH TO CRITICAL

The guiding principle followed in specifying the procedures used in bringing a reactor to the critical condition is that neutron flux levels shall be limited so that there shall be no damage to any of the reactor components. The rate of increase of flux must be limited as well as the highest level reached. The rate of increase of neutron flux is measured by the reactor "period", that is, the time interval during which the average flux density (power level) increases by a factor of $e = 2.718$. (Three "periods" yield a twenty-fold increase in power.)

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At the present time the approach to critical is made without positive knowledge of the condition of the reactor. The most sensitive instruments (aside from experimental prototype equipment at one of the reactors) cannot be guaranteed to indicate the condition of the reactor until the power generation rate exceeds 500 watts. At this time the reactor is on a rising period. Perhaps a minute is required before the value of the period may be estimated. If the initial period is longer than about fifteen seconds the reactor can be controlled manually from the operator's console. If the initial period is shorter than this the safety circuit will be manually tripped, introducing the safety rods and shutting down the reactor. The approach to critical may then be repeated in such a manner that undesirably short periods are avoided.

Safe operating procedures are relied upon to prevent undesirably short initial reactor periods. Loading patterns are designed to prevent the reactor from becoming critical before the safety rods are all out of the reactor. However, since the possibility of gross loading errors, or gross reactivity prediction errors does exist, the safety rods are withdrawn at a maximum rate of reactivity addition which will permit the reactor to be shut down before attaining the normal operating power level. Horizontal control rod withdrawal rates are designed to permit the operators to manually control the reactor at all times.

In the event that operating procedures fail to prevent undesirably short rising periods during startup, automatic instrumentation is provided to trip the safety circuit when the megawatt power level is reached during a cold startup, and at one-tenth operating level in the case of the hot startup. These instruments are reset to trip the safety circuit at higher power levels after controlled rising periods are established.

3.5 GENERAL CONSIDERATIONS

Since 1950 the increase in operating power levels for the reactors has been accompanied by a corresponding increase in the magnitude of reactivity prediction errors. These errors arise from increasing

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uncertainties in the values of the short term reactivity transients. A program of re-evaluation of the major components of the reactivity prediction calculation is currently being pursued. The principal uncertainties appear to be in the graphite and uranium temperature coefficients, with smaller uncertainties in the values of xenon effects. It is anticipated that the trend toward larger prediction errors will be reversed in 1957, and that by the time post CG-558, CG-600 power levels are attained reactivity prediction errors will have been reduced to values below those current for the present power levels.

Continuing programs for horizontal control rod calibrations will lead also, to better known reactivity balance during the approach to critical, and will thus reduce the probability of undesirably short periods during startups.

Instrumentation for monitoring the neutron flux during the approach to critical has been developed and tested. This instrumentation, including a low-level period-trip device, should eliminate the occurrence of undesirably short periods during startups. It is expected that this "sub-critical monitoring" system will be in operation by the time post-project power levels are reached.

Thus, although the short term reactivity transients will be larger and faster under post-project conditions, it is believed that the hazards associated with bringing the reactor to equilibrium power levels will be even less than at present.

If enriched fuel material were used to "flatten" the flux at the B, D, DR, and F reactors, which have only nine control rods each, a new problem might be introduced. The neutron flux level within the reactor at a given power level may become sufficiently unstable to cause "standing waves" of flux and temperature variation with cycling periods of several hours. Supplementary reactivity control by inserting "poison splines" between the ribs of the process tubes is under development. If flux (and thermal) cycling develops, and cannot be controlled by the "poison spline"

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method, prevention of cycling may be the controlling factor in determining the amount of fringe enrichment to be used in the older reactors. On the basis of existing graphite coefficient information, fringe enrichment for improving pile flattening is not contemplated at the old nine-rod reactors because of limited safety control capacity. However, it is planned that enrichment will be used to support "fringe poison" used as a shield protection measure; such use of enrichment is not expected to change the control capacity appreciably from that for natural uranium.

3.6 INSTRUMENTATION

The nuclear safety instrumentation system of a Hanford production reactor is required to cover a neutron flux range of 10^{10} to 10^{11} in the process of going from a normal shutdown to full power operation. That is, the full power flux may be 100,000,000,000 times greater than the shutdown flux. The background neutron source during a reactor "outage" consists primarily of photoneutrons produced by gamma radiation from the longer half-lived fission products impinging upon the natural deuterium present in the cooling water. The background flux is then proportional to the concentration of fission products, which, in turn, is dependent upon the full power neutron flux and upon exposure time. This means that the total flux range experienced in the normal operation of a reactor is essentially independent of the normal power level, and will not be significantly changed by projects CG-558 and CG-600.

The nuclear instrumentation for reactor safety and control is currently composed of four neutron flux level trip systems called "Beckmans"; a galvanometer system for fine control; and a proportional counter system for startup use. The relative effective flux ranges of the various operational channels along with several developmental systems is indicated in Figure 4. To the right of the chart the approximate fluxes in the reactor are indicated for the full shutdown condition, the region of criticality, the reduced level trips effective during cold and hot startups, and the full power flux.

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FIGURE 4

Effective Flux Ranges of Nuclear Instrumentation for
Reactor Safety and Control

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A startup occurring after the reactor has shut down and has passed through a reactivity minimum is called a cold startup. In this case the flux trip devices are set at full scale of the range on which the residual gamma induced signal is between three and 30 per cent of full scale. This establishes the trip in the neighborhood of one-thousandth of the projected full power flux.

A hot startup occurs after a reactor outage of short duration in which a reactivity minimum has not been reached. The flux trips, in this case, are set at one-tenth of the full power flux by adjusting the sensitivity range of the Beckman micro-microammeters.

During startups, while the reactor is on a rising period, the flux will over-ride the nominal trip setting as shown in Figure 5. Recalling that standard procedures require initial periods greater than fifteen seconds, it is seen that the flux over-ride during normal startups is essentially zero. In no case is it anticipated that the maximum flux attained will be greater than twice the nominal flux, and for the trip settings used even this over-ride would not be hazardous.

At full power levels, the flux trip is set at from ten to twenty per cent above the equilibrium level. For these settings the flux over-ride for a reactor disturbed from equilibrium would be negligibly small.

These instrument characteristics are discussed more fully in Appendix III. They will not be affected by Projects CG-558 and CG-600.

3.7 EMERGENCY SHUTDOWN SYSTEMS

3.7.1 Safety Circuits

The safety circuits of the Hanford production reactors are composed of three interrelated safety circuits, the 1X, 2X, and 3X. The 1X circuit consists of those relays and contacts which will release the vertical safety rods when deenergized. The 2X circuit controls the horizontal control rods in a similar manner. The 3X circuit is the last ditch, fast acting safety

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FIGURE 5

Representative Power Level Ratios as a Function
of Rising Reactor Period for Hanford Reactors

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system which releases neutron-poison balls into the vertical safety rod channels. The actual safety circuits differ a little from reactor to reactor; however, Figure 6 is a comparatively accurate representation for any reactor affected by Projects CG-558 and CG-600. The major components are tabulated below with the designations from Figure 6; the circuit in which it is found; and an indication of the usage. A is for the active, I is for inactive, a dash indicates the circuit does not exist.

Function	Desig.	CKX	B	C	D	DR	F	H
Beckmans - Four high flux trips which initiate scrams at excessive flux levels. By-pass interlocks are included to provide at least two active trips in circuit at any one time.	BX BXA	1XX	A	A	A	A	A	A
Dual area interties provide a scram at companion reactor if two or more Beckmans trip. There are two circuits - one outgoing and one incoming.	HX HXA GX-GXA	1XX	A	A	A	A	-	-
Panellit - A high and low pressure system sensing proper flow in each process tube. System has a contact type gauge for each tube. A ground detector senses grounding which might bypass a portion of the system.	PSR	1XX	A	A	A	A	A	A
Process water supply low pressure switches provide a scram if the water supply falls below a fixed minimum pressure.	LP LPA	1XX	A	A	A	A	A	A
The pressure decay system senses the rate of pressure decrease and drops the Ball 3X if the rate is over a fixed amount. System consists of two pressure switches and a time delay which bypasses the lower pressure switch and prevents the Ball 3X scram if the pressure has not decayed sufficiently to be hazardous in the time interval.	TD TDR LLP	3XX	A	A	A	A	A	A

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Function	Desig.	CKX	B	C	D	DR	F	H
The undervoltage system senses the control power at critical points in the pump houses and in the reactor building. A scram is provided for any condition which could cause pump or control failure.	PF PFA	1XX	A	A	A	A	A	A
The seismoscope system initiates a scram on the coincidence signal from two seismoscopes. The trip setting is at Modified Mercalli intensity 5 (just below threshold for minor structural damage.)	CX CXA	1XX 3XX	A -	- A	A -	A -	A -	A -
Intertie 1XX to 2XX circuit causes the horizontal control rods to be inserted if any 1XX scram occurs.		2XX	A	A	A	A	A	A
The VSR to 2XX intertie causes the horizontal control rods to be inserted if any vertical safety rod falls into the reactor.	SN	2XX	A	A	A	A	A	A
The 3XX to 2XX intertie causes the horizontal control rods to be inserted if any column of balls of the Ball 3X system drops. This automatically shuts the reactor down to prevent irradiation of the balls.		2XX	I	A	I	I	I	I

3.7.2 Response Time

The time sequence of events which are vital to reactor safety in the event of an emergency shutdown, or "scram", are illustrated in bar form in Figure 7. This chart is typical of C reactor; the principal difference at other reactors is in the length of time to insert the horizontal control rod system (up to twice as long). Time bar A is typical of a scram initiated by low process water supply pressure, a plugged or ruptured process tube, or a seismoscope trip. Time bar B portrays the conditions during a scram initiated from excessive neutron flux. Time bar C shows the action of the pressure decay system in tripping the Ball 3X circuit; a vertical safety rod and horizontal control rod scram should accompany this trip through the action in time bar A. Time bar D is typical of the scram

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FIGURE 6

Schematic Diagram of Reactor Safety Circuits

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FIGURE 7

Typical Time Sequence for Safety Circuits
(Data applies to C Reactor)

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resulting from undervoltage of critical power. There are variable time delays which may precede the undervoltage trip depending on how long a fault may be tolerated at any particular point without hazard to the reactor.

3.7.3 Reliability

The system utilizes the idea of dual relays and dual contacts at all critical points; the relay coils being in parallel while the contacts are in series. Examples of the dual relays in the foregoing discussions are the IXX and IXXA, the BX and BXA, and the HX and HXA relays. No arc suppression or damping circuits are allowed to operate across a contact as a short in the suppression circuit might bypass the contact; all such circuits are across the relay coils. All relays and contacts are designed with a very conservative approach so that the large design safety factors exist. Energy to insert the horizontal control system in the event of electric power loss is provided in the form of hydraulic accumulators at B, D, DR, and F and in the form of battery banks at C and H. The control rod system is the only part of the safety circuit which requires power to accomplish a scram; all other portions fail safe on a power loss.

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4. REACTOR COOLING

4.1 HEAT GENERATION RATES, AND LIMITING FACTORS

The increased water flow to be provided by Projects CG-558 and CG-600, combined with a higher tube outlet water temperature limit, will permit heat generation rates within the central process tubes to be increased to about 1.75 times their current value at B, D, DR, F and H reactors. At C reactor the potential heat generation rate will be about 1.17 times its current value. The heat generation rates to be actually attained may be limited by one or more of the following five factors: (1) Slug rupture rate, (2) Flow stability, (3) Graphite damage, (4) Corrosion, and (5) Biological shield damage. These factors are discussed below.

4.1.1 Slug Rupture Rates

The frequency of slug ruptures will probably impose the primary limitation to the reactor power levels attainable after the increase in water flow rates. With fuel slugs of present quality and with present goal exposures no significant power level increases can be made. However, if the goal exposures at B, D, DR, F and H reactors are reduced at the same time the water flows are increased, power level increases nearly proportional to the water flow rate increases will probably be realized without exceeding acceptable slug rupture rates. At C reactor, already operating with low goal exposures, no significant power level increases are possible until the rupture resistance of the fuel slugs is improved. A vigorous program designed to improve the stability of fuel slugs during irradiation is currently being pursued.

4.1.2 Flow Stability

Although heat generation rates are limited at the present time by the fuel slug rupture rates, it is possible that flow stability requirements will determine the limit as fuel stability is improved, or for low goal exposure operating conditions. Fuel slug damage can result from the

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excessive temperatures associated with steam formation and the resulting unstable, or variable, water flow. Current philosophy of operation requires shutting down the reactor before the onset of unstable flow conditions. This is accomplished by continuously monitoring each process tube inlet water pressure, with a "Panellit" gauge. High or low pressure on the panellit will automatically trip the number one safety circuit, dropping the vertical safety rods into the reactor and stopping the chain reaction. In addition the individual tube outlet water temperatures are periodically surveyed in order to assure that they remain below an established limit. Currently this limit from flow instability considerations is about 115 C, but following the increased water flows under projects CG-558 this limit may be raised.

4.1.3 Graphite Damage

Current heat generation rates result in graphite temperatures of about 500 C. At the higher power levels associated with the increased water flows, the graphite temperature will exceed 600 C, and may reach 650 C - the current graphite temperature limit.

Operation at the higher temperatures will not perceptibly shorten the life of the graphite stack provided the reactor atmosphere can be kept dry and relatively free of oxygen. However operation at higher temperatures will aggravate process tube curvature by increasing the graphite contraction in the central region of the reactor, while the current graphite expansion in the peripheral or "fringe" regions of the reactor remain unchanged. Over-boring of the process tube channels, and methods of increasing the relative heat generation rates near the front face of the reactor are being investigated as means of alleviating this situation.

4.1.4 Corrosion

Both the aluminum process tubes, and the aluminum jackets which isolate the uranium fuel slug from the cooling water are subject to corrosion in the reactor. The rate of corrosion is determined primarily by impurities in the water and by the temperature of the water. The current 105-C maximum tube outlet water temperature limit is established primarily by the necessity for limiting the slug rupture rate.

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As long as this tube outlet water temperature limit is continued, the total slug jacket corrosion for a given radiation exposure at the higher water flow rates will be decreased due to the reduced radiation time required for a given exposure. Furthermore, the planned reduction⁽¹⁾ in goal exposures will further reduce the time the fuel slugs will remain in the reactor. Consequently, jacket corrosion should not limit heat generation rates. However, as the maximum tube outlet temperature is allowed to approach 120 C, jacket corrosion might become a limiting factor.

Process tube corrosion rates are not expected to become a limiting factor under this program.

4.1.5 Biological Shield Damage

The increased heat generation rates under these projects is expected to increase the rate of biological shield deterioration. This is not expected, however, to become a limiting factor. Biological shield damage is discussed further in Appendix II.

4.2 HEAT TRANSFER

About 94 per cent of the heat generated in a Hanford reactor is generated by the fission process within the uranium fuel elements. This heat must be transferred by conduction to the surface of the uranium, through the slug jacket and into the cooling water, whence it is removed from the reactor with the water. Precise and detailed knowledge of these processes is not currently available for transient conditions. Consequently, conservatively large safety factors must be used in order to assure safe operation of the reactors, and the efficiency of production is reduced thereby. A continuing program of testing and analysis is in progress. This program is discussed in section 5.5.

4.3 CONTINUITY OF WATER SUPPLY

A continuous supply of cooling water, at an adequate flow rate, is absolutely essential to the safe operation of a Hanford reactor. Complete loss of inlet water pressure in a reactor operating at full power would induce

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the reactor to destroy itself within a few seconds, with consequent release of fission products to the environs. This would occur even though the nuclear chain reaction were stopped by the insertion of the vertical safety rods. The probable sequence of events which would follow loss of cooling water was discussed in section 2.2.

The continuity of water supply upon completion of projects CG-558 and CG-600, will be assured, in principle, by multiple pumping units with electrical primary power, and steam standby power; by multiple storage tanks, some of which are capable of supplying cooling water to the reactors by gravity flow; by multiple pipe-lines within each reactor area; and by pipe-lines interconnecting the reactor areas.

Continuity of water supply has been provided for the following emergency conditions:

- (1) Loss of electric power to all reactor areas simultaneously.
- (2) Simultaneous loss of electric and steam power at one area only.
- (3) Loss of steam power at one or more areas.
- (4) Loss of raw water.

Cooling water backup facilities have been designed to supply sufficient water to the reactor to limit the maximum process tube outlet temperature to 90 C, which is considerably below the current 105 C operating limit, and to provide sufficient excess water pressure to suppress boiling within the process tube which might be started by a momentarily inadequate flow. This protection is provided, except during the first three or four seconds after the beginning of a shutdown due to loss of electric power. The adequacy, of the water supply system under various emergency conditions is discussed below:

4.3.1 Loss of Electric Power

At each of the reactor areas, loss of electric power would stop the inflow of water to the storage tanks. The safety circuit would simultaneously shutdown the reactor and start the steam powered secondary pumps. The

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speed of the electric process pumps which pump water from the tanks through the reactor would decrease as the energy stored in the flywheel is expended. The combined effects of the flywheel momentum and the accelerating steam pumps will hold the rate of water flow decay to that shown in Figure 8, where curves are shown for pre- and post- CG-558 operating conditions. These curves are compared to the water flow rates required to prevent an increase in the effluent water temperature.

The response of the equipment at the C reactor area differs from that in the other areas mainly in that the steam turbines drive the regular process pumps rather than set of standby pumps. Water flow decay curves for the C reactor are shown in Figure 9.

The steam pumps can maintain indefinitely the shutdown flow required.

4.3.2 Loss of Steam and Electric Power Simultaneously, at a Single Area

In the event of simultaneous loss of electric and steam power in any area the safety circuit would shutdown the reactor, and the water flow would decrease as the flywheels expend their stored energy. About 1.7 minutes after power loss the decreased pressure in the main headers would cause the valves to the "high tanks" to open. Two of these tanks at each reactor contain a total of 600,000 gallons of filtered water. These tanks will provide a flow of 11,500 gpm for about 50 minutes at B, D, DR, F, and H reactors. At C reactor they will provide a flow of 16,000 gpm for about 35 minutes. When this supply is nearly exhausted, manually operated valves will be opened to introduce untreated water exported from adjacent areas into the reactor. Steam driven export pumps will supply adequate cooling water for the shutdown reactors even though electric power has been lost to all areas.

The water flow available versus the water flow required for B, D, DR, F, and H reactors is shown in Figure 10. Corresponding curves for C reactor are shown in Figure 11. Considering post CG-558, CG-600 operation, two seconds after simultaneous loss of steam and electric power,

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FIGURE 8

Water Flow After an Electrical Failure
(B, D, DR, F and H Reactors)

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FIGURE 9

Water Flow After an Electrical Failure
(C Reactor)

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FIGURE 10

Water Flow After Simultaneous Loss of
Electrical and Steam Power
(B, D, DR, F and H Reactors)

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FIGURE 11

Water Flow After Simultaneous Loss of
Electrical and Steam Power
(C Reactor)

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the water flow rate is sufficient to hold the temperature of the effluent at its original value. For the next hundred seconds a considerable excess of water flow is available to compensate for any temperature transient which might have been initiated during the first two seconds. Following the first hundred seconds, the high tank flow, and the import flow will hold the maximum tube outlet temperature below equilibrium value indefinitely.

4.3.3 Loss of Steam Power at One or More Areas

Upon completion of Project CG-558, loss of steam power at any area would not directly affect the process water flow. However, the reactor would be shut down until steam backup power was again available.

Currently, at B, D, DR, F and H reactors loss of steam power would cause an immediate process water flow reduction of about thirty per cent. This could result in severe temperature transients within the reactor. Consequently the possibility of reactor damage upon loss of steam will be significantly reduced by this change.

The C reactor is currently operating under these improved conditions.

4.3.4 Loss of Raw Water at a Reactor

In the event that the water supply from the river pumphouse is lost the reactor would be immediately shut down. Filtered water is available for at least twenty-four hours of shutdown flow. Water for at least another twenty-four hours cooling is available from the raw water reservoirs. This water may be filtered and used in the normal fashion if pumping power is available. If the river water supply is not restored by the time this water is exhausted, raw import water will be used to cool the reactor. This supply may be continued as long as raw water is available from other areas.

4.4 RELIABILITY OF WATER SUPPLY SYSTEM

The problem of water system reliability is a complex one, with many inter-related components and effects. A full analysis will not be attempted, but rather a short discussion with some conclusions concerning the relative

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safety of the water supply system after completion of projects CG-558 and CG-600 will be presented.

The basic philosophy followed in the design of the Project CG-558 and CG-600 water supply systems was that the assurance of continuity of flow of the revised system should be at least as good as that of the original system. This goal has been achieved in the design.

In several respects the reliability of the water supply system will be improved after completion of projects CG-558 and CG-600. The availability of one spare pumping unit, on standby, will make possible the loss of one operating unit with essentially no interruption of the process water flow. Loss of standby steam pressure will not directly affect the process water flow. The increased process tube water pressure, which will result from increased flow rates with unchanged rear face tube nozzles and pig-tails, will decrease the probability of unstable flow at a given tube outlet temperature. The new cascade type downcomers at B and D reactors will increase the rear header pressures and permit higher tube outlet temperatures without the initiation of boiling in the rear headers. The major front face piping will have sufficiently greater strength to yield a significant increase in the ratio of design stress to operating stress, thus having a greater safety factor upon completion of project CG-558. And, finally, the installation of check valves in the front face crossheaders will permit one riser or main header to be ruptured without causing loss of cooling water to the process tubes.

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5. DEVELOPMENT PROGRAMS

5.1 HAZARDS ASSOCIATED WITH UNCOOLED REACTOR

A comprehensive program to determine the hazards associated with an uncooled reactor has been undertaken. Initially, efforts are being concentrated on determining the quantity and rate of release of fission products from irradiated uranium being oxidized in air at various temperatures. The effects of uranium-aluminum diffusion, and the presence of graphite, on the fission product release will also be studied. The effectiveness of filtering the reactor building exhaust in preventing spread of contamination is currently being studied.

5.2 REMEDIAL ACTION IN THE EVENT OF LOSS OF COOLING WATER

The emphasis at HAPO has been, and is, primarily directed toward preventing loss of process cooling water, or similar disaster. However, a program has been initiated to determine what action could be taken to minimize and to delay the release of fission products after loss of process water has occurred.

The discussion of remedial procedures is complicated by the great variety of different conditions that could exist, and by the uncertainties in the time scale and the magnitudes associated with the events following water loss. However some general conclusions may be reached:

(1) The reactor should be shut down as quickly as possible at the first warning of possible water loss. All available poison control should be inserted into the reactor. This will reduce the power level at the maximum rate, increase the time for remedial action, and reduce the probability of the reactor becoming super-critical after melting of uranium.

(2) The reactor should be cooled as much as possible. If any process water is available its use should be continued. Even partial cooling of the reactor will reduce the temperature rise and thus reduce the probability of the reactor again becoming super-critical. Furthermore, peripheral cooling will help contain fission products released in the central zone of the reactor.

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(3) The process water should be restored as quickly as possible. If the interval without water indicates that extensive slug damage has occurred the flow rate should initially be limited to that rate which will remove the fission product decay heat as fast as it is being generated. In no case should the water flow rate be sufficient that steam pressure buildup within the reactor core could topple the shields and blow the roof slabs off the reactor building. To minimize the water borne contamination, and prevent possible overloading of the cribs, the flow rate should be reduced as rapidly as consistent with the temperature and heat generation rates within the reactor.

(4) If the release of fission products cannot be prevented, it should be minimized. Prior to abandoning an area, the reactor atmosphere and the reactor building ventilation systems should be shut down. The interior and exterior doors of the reactor building should be closed.

5.3 SUPPLEMENTARY REACTIVITY CONTROL

Two supplementary systems for controlling the reactivity of the Hanford reactors are being developed. The first of these, a "Poison Spline" system is for operational control, and the second, a boron tri-fluoride injection system is designed to assure that the reactor will remain sub-critical after being shut down due to water loss at full power, or other similar disaster.

5.3.1 The Poison Spline System

In the poison spline system, strips of neutron absorbing material would be introduced into selected process tubes beneath the fuel slugs and "between the ribs" of the process tube. This is to be accomplished while the reactor is operating at full power, by the use of modified frontface nozzle caps. This system would be used primarily to increase the operating efficiency of the reactors. It could also be used to reduce the reactivity during shutdown to assure that the reactor would remain sub-critical in the event of water loss, and to compensate for long term reactivity gains.

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5.3.2 Boron Trifluoride Injection System

In the event of a disaster such as loss of cooling water while the reactor is at full power, or a power excursion which results in boiling the water out of the process tubes, the uranium fuel slugs would melt within seconds. Xenon poison would be released from the metal and would gradually escape from the reactor. As seen from Table III of section 3.3, loss of a large fraction of the xenon from the B, D, DR, F and H reactors could cause the reactor to again become critical. More fission products would be generated, and the severity of the disaster would be increased. The C reactor would probably remain sub-critical but even this is not certain.

Since the reactor would be unusable following a disaster in which a large fraction of the uranium is melted, further damage to the reactor is not a factor in providing a "last ditch" supplementary reactivity control system. The system currently being developed consists of a boron trifluoride storage tank together with a special connector between the storage tank and the reactor atmosphere system. This connector is expected to be a section of pipe which must be manually installed before the system can operate, or some similar "missing link" type of connector. Suitable capacitance type alarm systems and key locked controls would be provided to prevent inadvertent use, or deliberate sabotage.

A program of laboratory work is currently being carried on to determine the amount of boron which would be retained by the graphite under the conditions expected to be found in a reactor following loss of cooling water. Results of this program indicate that poisoning equivalent to several thousand inhours of reactivity would be retained on the graphite following two or three hours exposure to boron trifluoride gas at temperatures between 600 and 1100 C. Work in progress is expected to yield (1) the effect of temperatures up to 2000 C in reducing the amount of boron trapped by the graphite; (2) the effect of water vapor in the reactor atmosphere, or moisture on the graphite, on the amount of boron retained by the graphite; and (3) the effect of oxidation of the graphite on the boron content.

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5.4 NUCLEAR INSTRUMENTATION

There is a continuing program of improving reactor safety through better instrumentation. This program is aimed primarily toward covering those regions that are not completely monitored at the present such as the sub-critical neutron flux, the initial post critical rising periods, and tube outlet temperature during transient conditions. A secondary function is that of improving and expanding existing systems.

5.4.1 Sub-Critical Flux Monitor

The Sub-Critical Neutron Flux Monitor⁽¹⁾ is a system that is capable of recording the neutron flux when the reactor is in the full shutdown condition. This is a pulse type of system composed of a U²³⁵ fission counter, a remote positioning mechanism, preamplifier, amplifier, logarithmic counting rate meter, period amplifier, and a trip, a four decade recorder, and a positioner control. The remote positioning mechanism is such that the fission counter may be inserted into the fringe region of the moderator during shutdowns so as to monitor the very low neutron flux. As the reactor is made critical and is brought up to the operating level, the fission counter is withdrawn, in steps, to a protected position within the reactor shield. This action gives the system a very wide operating range and protects the counter from burnout and radiation damage.

The Sub-Critical Monitor is now in the prototype stage. It is used in observing the change in multiplication of the background neutron flux as the vertical safety and horizontal control rods are withdrawn during the approach to the critical condition. Such information has its greatest value in confirmation, or correction, of the calculated critical conditions and in its support of the established rigorous startup procedures employed at Hanford.

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Absolute reactivity values cannot be obtained directly from the counting rate, or neutron flux at one location in the reactor, because of the flux unbalances that exist under extraordinary reactivity conditions and because of the dependence of the neutron background upon the previous operating history. The system as now developed is adequate while remaining relatively simple.

The period amplifier and safety circuit trip portion of the Sub-Critical Monitor is used when the reactor is sub-critical, and also during the initial power rise. The trip will assure that criticality is established with a controllable period longer than 15 seconds, which is the fastest allowed under the Process Standards.

5.4.2 High Level Period Trip System

A second type of period amplifier system, the so-called High Level Period Trip, is undergoing development. This system consists of a standard gamma compensated neutron sensitive ion chamber, a six decade logarithmic DC amplifier, period amplifier and trip, and recorder combination. The development in this case has been more along the lines of application and utilization. This system will be used to control the progress from the kilowatt range to full power, in place of current practice of range-switching a by-passed linear Beckman flux trip channel. One such system is currently being operated at KE reactor.

5.4.3 Octant System

An effort is currently underway to convert the galvanometer systems at all reactors into octant type systems wherein eight chambers are installed at the corners of the reactor block, as at C reactor. This will be accomplished by placing commercial ion chambers in existing facilities at DR and H. The B, D, and F reactors have no such facilities and a development program is under way to provide a gamma compensated neutron sensitive ion chamber of small enough dimensions to be inserted into the front and

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rear ends of four corner process tubes. One such ion chamber is currently installed and being tested. Initially these chambers will be operated in an octant type of galvanometer system. It is then planned to expand the system with the addition of safety circuit flux level trips from each of the chambers or in sets of two so as to distribute the trips about the reactor rather than having them concentrated beneath the reactor as at present. Ultimately this system will probably replace the present Beckman systems in providing the full power flux level trips.

5.5.5 Outlet Temperature Monitoring and Display

A project, CG-666, is currently underway for installation of safety circuit process tube outlet temperature monitoring at all reactors exclusive of KE and KW, which already have such systems. This system will be a zone or cell type of temperature monitor operating on 211 process tubes. Each of the monitored tubes will be located in a zone or cell, usually of nine tubes. The design will be such that, with an allowance for the temperature differential between the control tube and the hottest tube in the cell, the reactor will be shutdown if any process tube exceeds a specified temperature. The temperature monitor will include an approach-to-trip panel on which a lamp will light for each control tube that is within a particular temperature range of its trip as determined on a master approach-to-trip scan control. In this manner incipient trips may be located and corrective action taken before a scram results.

The temperature monitor system will have its strongest effect on reactor safety in monitoring the heat shifts that occur under transient conditions such as during scram recoveries and periods of rapid rod motion controlling reactivity transients following startups. The existing temperature data transcribers require about ten minutes to obtain a full temperature map, and the conditions may change considerably during this interval. The temperature monitor will permit safe operation much closer to limits under these transient conditions.

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After the development of new reactor safety instrumentation to regions which are currently controlled by specified procedures, the procedures will continue in force with the instrumentation to act as an enforcement agent.

Other instrument development efforts currently being pursued are aimed at improving such items as the panellit pressure monitor, the ion chambers, the counters, ruptured slug location, and data display.

5.5 HEAT TRANSFER

A continuing program of testing and analysis is in progress. This program is designed to yield more complete and accurate knowledge of the heat transfer characteristics of the reactors. This knowledge will permit more efficient operation with equal or greater safety, and will permit better estimates of reactor behavior under emergency condition such as a power excursion or various degrees of cooling water loss. The main components of this program are discussed below:

5.5.1 Laboratory Program

The laboratory is provided with full scale process tubes attached to inlet and outlet headers by process type nozzle assemblies so that flow conditions within the reactor may be accurately reproduced. Prototype nozzle assemblies, and special tubes, may be installed for study. Fission heat generation is simulated by electrical heating units enclosed within an aluminum or Cu-Ni alloy tube having approximately the same outside diameter as the standard fuel element. Four programs are currently being pursued in this laboratory:

The first of these is to determine the conditions that lead to boiling within the process tube and the fuel element surface temperatures associated with this boiling. The whole picture of the undesirable effects of boiling and the limiting conditions that determine fuel element damage is unclear. It is believed that this study will lead to safer operation of the reactors at higher power levels.

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A second program is designed to establish the correlation between the pressure monitor (Panellit gauge) trip settings and the initiation of unstable flow for various heat generation rates and process water flow rates. Unstable flow is the name given to any water flow rate which is less than the minimum continuous flow required. It is a varying flow rate in a process tube caused by formation of steam in the tube. The rate of formation of the steam must be sufficient to reduce, but not large enough to reverse, the rate of water flow into the process tube. Closer examination of these phenomena may show that the current "trip before instability" concept is unnecessarily conservative, and that a "trip after instability", that is, tripping the safety circuit after the initiation of unstable flow, would be sufficient to prevent damage to fuel elements. This would permit maintaining higher tube power levels, with a corresponding increase in production rates.

A third program, being evaluated, would be designed to determine the effects of boiling in the rear crossheaders upon process tube flow. The rear header in a reactor has forty-six process tubes feeding into it, while the flow laboratory rear crossheader has, at most, three. Knowledge of the effects of rear crossheader boiling is essential to the proper evaluation of results obtained from single, or few, tube mock-ups.

The fourth program is a general study of the flow stability for proposed fuel element designs and the effectiveness of pressure monitoring systems in maintaining stable flow for these designs. Currently, the "I and E" slug, a hollow fuel element which would be internally and externally cooled, is being studied. The present system of flow control by monitoring the tube inlet pressure with a Panellit gauge would be relatively insensitive to partial blockage of either the internal or external flow channel. Consequently, the changes in the flow monitoring system and in the tube specifications which will best provide adequate protection to the fuel elements are currently being determined.

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5.5.2 Combined Laboratory and In-Reactor Program

A series of tests are planned which are designed to determine fuel element temperatures as a function of reactor power level; fuel element design; water flow; the amount and location of adjacent poison such as control rods, or "flattening" columns; and various other parameters in order to assure continued operation within safe temperature limits. At the same time these tests should yield valuable information concerning possible thermal stresses in these fuel elements, and the effects of irradiation upon their thermal properties.

5.5.3 In-Reactor Program

A study will be made of the temperature transients produced in a reactor by various emergency conditions, and of the parameters which determine the magnitude of these transients. At the present time, for example, no measurements have been obtained of the temperature transients which follow loss of electric power from the Bonneville Power Authority. Following loss of electric power, the flywheels on the electric motors will continue to drive the primary pumps at a continuously decreasing speed, the steam plants will automatically accelerate to full power within perhaps twenty seconds, and, if the header pressure drops below a predetermined value, water will flow to the reactor from the high tanks. The prevention of a disaster is thus assured. However, there is a period of two or three seconds during which the water flow rate is insufficient to remove heat at the rate at which it is being generated in the reactor. This might be sufficient to initiate a severe temperature transient within the reactor.

This temperature transient will become more severe as the power level of the reactor is raised. In order to assure safe operation of the reactors at post CG-558, CG-600 power levels, this temperature transient will be evaluated, probably by extrapolation from data obtained from simulated loss of electric power at reduced power levels.

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5.5.4 Computer Program

An electric differential analogue computer will soon be available at Hanford. This computer will be used to determine probable effects of various catastrophic conditions in the reactors. As more complete calculations can be made, and these supplemented where possible with experimental tests, our knowledge of safe operating limits and the probable results of exceeding these safe limits will be considerably increased.

5.6 SUPPLEMENTARY COOLING

In spite of the effort expended to assure continuity of cooling, certain foreseeable conditions could result in cooling system failure. These include: violent earthquake, enemy bombing, sabotage, and the flood which would result from the destruction of the Grand Coulee Dam. An earthquake, blast, and flood resistant supplementary cooling system might prevent destruction of the reactor, and, failing this, could greatly reduce the release of fission products, in the event of complete loss of the primary and backup water supply systems.

It has been impossible to recommend a supplementary cooling system for the existing reactors because information necessary for the evaluation of the economic incentive for preventing a reactor disaster as a result of an act of war has not been available. Unsuccessful efforts have been made by the Atomic Energy Commission to develop information applicable to this problem. Consequently, the study of supplementary cooling systems useful primarily in the event of acts of war has not advanced beyond preliminary scoping for cost estimating purposes.

A similar situation exists with respect to earthquakes. There is not sufficient information available to permit estimation of the probability of a severe earthquake occurring in the Hanford area during the lifetime of the six reactors considered here. The probability appears to be small, but not negligible.

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HAZARDS SUMMARY REPORT

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