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TECHNICAL STUDIES SUPPORTING IPD DISASTER PLANNING

AUTHOR P. A. Carlson

R. T. Jaske

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TECHNICAL STUDIES SUPPORTING IPD DISASTER PLANNING

August 5, 1963

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IRRADIATION PROCESSING DEPARTMENT

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DECLASSIFIED**PREFACE**

When the Atomic Energy Commission request for this study was received, the authors were requested to contribute technical support to final recommendations forwarded to the Richland Operations Office. This joint report establishes a formal summary of the information furnished as a basis for requested project action.

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TECHNICAL STUDIES SUPPORTING IPD DISASTER PLANNING

1. INTRODUCTION

At the request of the Richland Operations Office a study of means to render the plant less vulnerable to self-destruction with accompanying spread of radioactive contamination was undertaken. A basic ground rule imposed required that the suggested measures be effected immediately and with little capital cost. An examination of available alternatives indicated that within the budgetary and time limitations imposed by the request, no significant period of unattended operation could be guaranteed. Accordingly, it was concluded that the only course of action available was to totally discharge the reactors and provide water cooling flow by gravity to the reactor storage basins.

It is the purpose of this report to review the technical considerations involved in determining the feasibility of the selected course of action with specific emphasis on the problem of assuring basin integrity. Alternative course of action will not be discussed, having been covered in the official correspondence and reference reports. Ref. (1)(2)(3)

2. SUMMARY AND CONCLUSIONS

The major problem to be overcome in selecting a basis for the proposed gravity system was a determination of the upper temperature limit and maximum feasible temperature differential to which the basins could be subjected. Having determined an approximate level, the available water resources were balanced against this limit and the heat load normalized for projected operating conditions. Secondary aspects such as water circulation and usage effectiveness, heat transfer from the discharge mass, drainage and vapor venting formed the balance of the investigation. A decision to use sensible heat gain for heat renewal rather than boiling formed a part of this phase of the work and was culminated in a series of curves showing required flow rates as a function of time. (Figures 1, 2 and 3).

Based on this supporting investigation, the authors conclude the following:

-
- (1) HW-75779 - Minimizing Consequences of an Attack Threat, C. A. Priode, December 7, 1962.
 - (2) HW-72563 - IPD Defense Capability Study Plan, R. T. Jaske, et.al., February 15, 1962.
 - (3) HW-75802 - Stabilization of Hanford Plant for the Purpose of Minimizing Consequences of an Enemy Attack, L. L. German, December 14, 1962.

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- A. No significant increase in leakage rate will accrue by virtue of heating the basin to approximately 80 degrees C. Cracking will occur on the cold side during the heating cycle and later, on the water side upon cooling. Upon recovery to normal temperatures, considerable cracking and spalling can be expected. However, it is not considered probable that any damage would occur which cannot be repaired, or which at any point in the heat cycle would create a leak in excess of the flow capability of the proposed gravity system.
- B. The transfer of heat in sensible form by the flow of coolant offers substantial advantages over a system which permitted boiling. A sensible cooled system minimizes the temperature to which the basin would be heated, reduces the corrosion effect on the fuel jackets, largely eliminates the vapor condensation problem on the rear face, and most importantly, provides for an excellent margin of safety should some leaking of the basin structure develop. The compensating advantage of lower project cost and longer flow time for a boiling system did not appear to be worth the extra risk within the ground rules established for the study.
- C. Since enriched material would be discharged first and appropriately segregated, so long as surface wetting of all discharged material is maintained it is not probable that any significant release of contamination will occur from the discharge mass. No addition risk would be involved if all normal discharged material were not fully bucketed providing minimum requirements of shielding and water circulating path were met. As a matter of good judgement it is suggested that operating plans include bucketing of all material in order to best use available water at lowest local fuel and basin temperatures.
- D. Records for discharged material using goal exposures for FY 1962 were used as a basis adjusted to a normal distribution of fresh and decayed inventory and carried at maximum power levels of 4850 MW at 105-K, 2500 MW at C, and 2200 MW at the five oldest reactors. This approach was considered prudent in view of the potential power level increases possible as a result of current plant programs.
- E. At the conclusion of the emergency period of from 17 to 20 days, prompt action to restore additional water flow would be required to ensure protection from release of contamination. A major factor in planning the restoration of activities would be the speed at which inventories could be worked off or shifted in order to permit 105 building rehabilitation and basin repair. Accordingly it is concluded that a period of some six to nine months could be entailed in any restoration of the plant to full normal activity based on current separations capacity, rail shipping capability and potential basin repair work following an actual evacuation.

Section 3.10 - Technical Basis for Gravity Water System Project

3.10 Background Information

Upon receipt of the Commission request for a review of the possibilities of measures which could be adopted immediately and with little capital cost, previous work was studied for guidance. Whereas in earlier requests, the resulting plan was derived from a basis of little or no advance warning, the present case was formulated around the availability of appropriate warning to enable a low-cost solution to be practicable, document HW-75779 (Ref. 1). No other clearly feasible alternatives which do not involve the requirement for operator attention and investment for shelters were developed. In view of the continued reluctance on the part of the AEC to provide funds for shelter construction, plans involving shelter facilities currently do not appear to be feasible within the foreseeable future.

Concurrently with this effort, a budget data sheet covering original intent of the department study plan (Ref. 2) has been forwarded for consideration in FY-1965. These funds would cover installation of shelters, oil conversion and general provisions to permit securing of the plant within minutes notice of impending attack. Since the fate of this proposal is subject to yet undetermined policy decisions by the appropriate agencies involved, it does not appear that a review of it or other alternatives is useful in this document.

3.11 Inventory Considerations

The key factors governing success of the proposed project is the development of an appropriate basis for the cooling requirements and minimum discharge times. These are, to a large extent, affected by Commission requests for non-representative plutonium production, tritium production or production related to space isotopes. In order to allow a study effort to proceed, these variables were eliminated by the assumption that no program would produce a heat release requirement in excess of a normalized weapons grade plutonium program at power levels anticipated with current R & D programs as shown in Table I. These levels are five to ten per cent higher than current administrative power limits.

TABLE IReactor Power Levels for Planning

<u>Reactor</u>	<u>Power Level, MW</u>	
	<u>Winter</u>	<u>Summer</u>
B,D,DR,F,H	2200	1833
C	2500	2083
KE, KW	4850	4040

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A remaining important factor is the need to bucket all enriched material in order to conform to criticality requirements. This is of special significance for loads such as the E-N load at 100-H and for disposition of the normal enrichment. In deriving discharge times, enriched material would be handled first and stored in a critically safe manner before effecting the normal uranium discharge.

Table II represents a summary of the inventory conditions assumed at the time of discharge. The K reactors are considered to contain Zircaloy-2 process tubes and, thus, KVN fuel elements. Based on these figures, discharge times were computed and budget requirements determined. A net deficit of buckets exists in any area with any extensive E-N loading; however, because of the uncertainties in planning for continued loads of this type, no provision for procurement of additional buckets is proposed as part of the project. Upon completion of the detailed planning and preparation of operation duty cards, any deficit of operational type tooling or equipment will be made up from normal expense funding.

TABLE II

Normalized Inventories & Exposures for Planning

	<u>In Reactor</u>		<u>In Basin</u>	
	<u>Natural</u>	<u>Enriched</u>	<u>Natural</u>	<u>Enriched</u>
<u>Inventories (Tons)</u>				
Each K	380	80	525	115
Each old reactor	185	35	200	40
H reactor (E-N)	4	200	6	220
<u>Exposure* (MWD)</u>				
Each K	350	415	660	830
Each old reactor	330	450	700	900
H reactor (E-N)	300	450	600	900

*The average cooling period of the metal in the basins was determined to be 77 days at the six older reactors and 68 days in the K reactors.

TABLE IIICURRENT STATUS OF BUCKET CAPABILITY

	<u>Available Buckets</u> ⁽¹⁾	<u>Buckets Required</u> ⁽²⁾	<u>Margin</u>
B	546	510	30+
C	605	510	80+
D	644	510	100+
DR	514	510	--
F	516	510	--
H	566	480	80+
KE	1656	1100	500+
KW	1464	1100	350+
CPD	<u>100</u>	<u>100</u>	<u>--</u>
	6670	5330	1140

(1) Estimate based on sample distribution taken in 1961 - does not account for unservicable galvanized buckets.

(2) All enriched metal requires stainless buckets for Redox process. This estimate does not cover distribution of buckets by type.

3.12 Discharge Times

Prior to development of a detailed operating sequence some aspects of the time of discharge remain somewhat speculative, however, in general the values arrived at represent best effort estimates derived from operating experience.

As stated in Ref. (1), the discharge operation would proceed as follows: "The reactors would be discharged in an orderly manner in accordance with process standards. The enriched fuel would be discharged first, bucketed and stored according to process standards. The remaining natural fuel would be discharged by flushing or by using the ram seater. The amounts of natural material that would be moved from or left in the chutes would depend on individual reactor chute capacity. Considering the reactor loads, types of fuel and chute capacity, the old reactors could be discharged in 7 to 10 shifts and the K reactors in 11 to 14 shifts.

"With the enriched fuel stored and maintained according to process standards, poison would be unnecessary in the natural fuel storage geometry. The total load including enriched and natural metal is not discharged together in one massive stack because of the difficulty of getting proper poison in solution would be dissipated by water flows and basin leakage (sic)."

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While covered in the section 3.33 on Circulation and Stratification, it is important to point out the undesirability of leaving any significant amount of metal in the chute. Such a pile would, to a great extent, result in a volume of metal which was inadequately cooled even though all of the fresh cooling water is introduced at the rear. This might occur because the hot water would remain trapped in the upper portion of stratification, since the only exit for the water addition is downward by density current flow. See Fig. 4.

This effect would be most pronounced at 105 B-D-F and perhaps the Ks, and would be of concern primarily to avoid a potential release of steam or fission products to the rear face which would cause delay in starting up again, either because of contamination or vapor damage.

While final plans remain for the future, the authors suggest that, depending on the nature of the alert, every effort be made to remove substantially all metal from the rear face chute area prior to actual evacuation.

3.13 Recovery

The time required to restart the reactors after an all-clear signal is basically a function of the capacity of the canning operation and time required for any necessary basin repairs. If the all-clear signal occurred immediately after discharge, one K reactor and one old reactor could be charged and restarted in about a week by supplementing the canning operation's production with finished fuel from normal fuel inventories. With fuel being fabricated, delivered, and charged daily, it is estimated that all reactors could be refueled and restarted, at the rate of one reactor per week in about seven to eight weeks. With an all-clear signal coming seven weeks after a false alarm occurred, the eight reactors could be refueled in ten days or less. However, the potential for damage to the storage basins must be considered. Even though the basin flows are adequate to preclude nuclear safety difficulties in areas shut down and unattended, some damage from heat cycling can be expected. If the alarm were determined to be false at an early date and regular basin flows were maintained, the start-up of reactors could proceed quickly following refueling. If personnel were evacuated for some period of time following the discharge of the reactors and heat cycling occurred in the basins, a period of six to nine months could be required, depending on the extent of the damage, to progressively restore all the reactors to normal operation.

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3.20 Thermal Studies of Basin Structures

At the onset of this investigation it was soon determined that a rigorous solution of the stress and expansion problem, while desirable, would a) not be available within the time allotted for the investigation, and b) may not be capable of theoretical representation without extensive supporting research. Since the expected occurrence of the evaluation plan was reasonably remote it was deemed prudent that guidance be obtained from the examination of structures subject to cyclical hot water duty over a period of time. In determining expected damage this examination revealed that for the most part, primary damage to concrete structures from hot water has been chiefly caused by gross motion of inadequately anchored appurtenances, primarily pipe lines. Where other damage was evident from thermal action it was in the form of cracking, subsidence from undermining and spalling in structures subjected to repetitive heat cycling over a period of ten to eighteen years. It appeared appropriate to assume that short of a catastrophic collapse, which was not deemed credible, any damage would be of a nature which would not compromise the basic function of the basin.

3.21 Thermal Stress in the Basins

The need for study of severe thermal action in concrete has been accelerated by the development of large dams and massive structures, atomic energy applications and rocket testing and launching facilities. Generally the literature coverage is spotty or consists of cases where confirming tests are impractical, or form a part of the classified computations supporting specific projects..

Designers in the nuclear business have until lately pioneered in applications involving high thermal gradients and internal heat generation although little material is still available. H. S. Davis has surveyed the situation generally in reference (4) which provides theoretical basis for this discussion.

Since the application of heat in this case is of short term nature it is not necessary to consider basic changes in concrete properties at elevated temperatures. Such changes, generally in the direction of dehydration and loss of strength are not ordinarily pronounced under 500 degrees F and of relatively low significance under 300 degrees F.

Of significance are:

- a) thermal stresses induced by thermal gradients through the affected slabs, and
- b) The stress induced by relative motion of portions subject to uneven heat gradients in one or more directions.

(4) Thermal Considerations in the Design of Concrete Shields, H. S. Davis, A.M. ASCE, Proc. Paper 1755, September 1958.

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Ordinarily, linear temperature gradients in a homogeneous body would not produce thermal stress unless the movements of the body are restrained. Since a concrete wall is always restrained internally to some extent and externally by design, some degree of stress can be expected from even nominal gradients.

To the extent that resistance to deformation occurs, the slab in question will tend to develop compressive stress on the hot side and tensile stress on the cold side. Ordinarily a nominal amount of reinforcement, 0.3 per cent or less, is used to control random cracking from minor temperature fluctuations and shrinkage. Where gradients become more severe additional steel is required. To the extent that elastic conditions are still followed, tensile steel can be added on the cold side. As soon as cracks and elongation appear, the basis for an exact calculation is absent, consequently it is common to assume as a computation basis that no stress cracking will appear. Such an assumption is only justified if absolute leak-tight integrity is required, say for a critical structure containing noxious gases or fluids. Ordinarily it is easier, as in the case of nuclear reactors, to apply a thin steel envelope and expect minor, well distributed cracking to relieve unbalanced thermal stress.

Another factor to consider is the buildup of moments within rigid frames. In new design it is common to offset any additive or subtractive moment from temperature with appropriate distribution of reinforcement. Depending on the frame to be analyzed, specific changes in length from expansion of components usually determine the resulting treatment. It is of concern that when a thin wall is connected to a relatively thick wall that thermal moments in the thinner member can exceed the fixed end moment which the section can resist.

In the case in point, the basin walls are cantilever structures designed to resist earth pressure without water or conversely, water pressure without backfill. This provides a good deal of extra margin since neither condition is expected to exist during the action in question. That portion of the reinforcement designated as temperature steel is nominally on the order of 0.1 per cent. Computations assuming theoretically fixed ends indicate that the yield of the available temperature steel would be exceeded by a gradient of from 30 to 40 degrees C. Theoretically, some 80 to 100 degrees C could be handled by an increase to 1.6 per cent reinforcement. Again in the actual case, using the extra credit from the balanced earth and water loading, it appears reasonable to assume that upwards of 0.8 to 1.0 per cent total reinforcement is available to offset thermal stress.

While the importance of exceeding the yield on the cold side is of theoretical significance it does not appear to be a controlling factor in the case in question. Study by E. F. Smith of CE&UO reported to the author (5) indicates that no significant increase in leakage would occur by virtue of

(5) Personal communication E. F. Smith to R. T. Jaske, February 14, 1963.

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bulk water temperature rising to 180 degrees F. He cautions that upon return to the normal temperature some cracking and spalling damage could be expected, but due to the indeterminate nature of the problem it is not possible to define the exact extent. It is probable, however, that the working of expansion joints would require some repair work to minimize future serious damage from subsidence. This latter area poses a greater long term threat than the temperature stresses per se.

Accordingly, the authors conclude that assignment of a temperature limit of 80 degrees C is prudent for the case assumptions. Values lower would be desirable and higher values less desirable. Using 80 degrees C as a first approximation it was determined that sufficient water was available; consequent work did not require an adjustment.

3.22 Local History

As a reference guide, structures within IPD subject to thermal cycles of the same order of magnitude were reviewed in order to determine the applicability of the line of reasoning adopted in selection of 80 degrees C as a design basis. Structures considered were:

- A. 107 B-D-F-DR-H Basins
- B. 1904 H and Spillway
- C. 1904 B-D-F
- D. 105 C-H Buildings (downcomer walls)

A. 107 Basins

While not all of identical construction, the basic concept is that of a gravity wall dam with a cast in place gunited floor. The newer basins, DR and H place greater emphasis on reinforcement and appear to have suffered less damage. Despite the great number of thermal cycles, the worst cast (107-F) when scaled down to a basin the size of 105, would leak less than 100 GPM, only some 30 per cent of the available flow. A great deal of spalling has occurred in the entry box of 107-F which has total reinforcement approximating the lower thicker section of the 105 basins. This damage while severe, is still not sufficient to require immediate replacement.

B. 1904-H Spillway

This structure originally completed in 1949 with reinforcement not too different than the 105 basins, has undergone successful service under thermal cycles since that date. In 1960 a repair program was necessitated by the thermal creep of the effluent lines (encased in concrete) pulling the on-shore end through and out of the box with consequent concrete damage which was repaired by installation of pipe sleeves and grouting. In 1961 the effluent lines plugged from motion of the river bed following completion of Priest Rapids Dam.

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For a period of several weeks and temperature cycles the overflow structure served as the primary effluent conduit. Inspection after the lines cleared themselves indicated only nominal cracking had occurred and full availability for service remained. The only obvious damage was the spalling of a small wall edge approximately 30x6x6 inches which had no thermal reinforcement. This piece still lies on the river bank and is available for inspection at any time.

C. 1904 B-D-F

These structures, somewhat like 1904-H, are simpler in scope and detail. Examinations during 1959 following fourteen years of service, showed minor spalling at D area and serious damage at F area. Here, as in H, the effluent lines had crept through their encasement toward the river and broken several large pieces of concrete loose. The subsequent tumbling of these pieces had broken through the box wall in one or more locations. The box was grouted and replaced in service without subsequent need for attention during a three day outage.

D. 105 Buildings

In the DR, H and C buildings, and later B, D and F, the downcomer is cast in place within four foot thick concrete walls containing approximately 0.15 per cent reinforcement. The DR and H buildings which have had the most service, show vertical cracks through which water has run when the liners were leaking. No serious cracking or spalling is noted beyond these cracks which leaked very little in volume with the 20-30 foot head available. The F building during the time when the downcomer liner leaked, dripped water through laitence in horizontal joints but has developed no obvious cracks.

It appears correct to conclude from this experience that, providing minor cracking and surface spalling are of little concern, and provided that absolute leak tightness is not required, concrete basin structures adequately reinforced for backfilling and hydrostatic loadings offer a relatively good chance of accepting thermal expansion up to 90 degrees C repeatedly over a long period of time, provided the walls are relatively uniform in thickness and do not adjoin thicker asymmetric structures.

3.23 Review of Other Work

While no extensive history of parallel investigation was discovered, a somewhat analogous situation was reviewed during detail design of 100-N Area. The study was occasioned by need to develop a reservoir for an emergency dump of the cooling loop water. Its temperature would approximate 212 degrees F after flashing to atmospheric pressure.

"The basin was not designed to withstand temperature transients of 60 degrees F/minute nor was it designed to take 180 degrees F or higher final temperatures. Cracking of the concrete walls and spalling of the surface are probable. Preliminary studies indicate cracking of a nature that would require repairs would be likely during each dump. While this might be acceptable for an actual dump,

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it would make system tests impractical and prohibitively expensive."(6)

Despite the obvious opportunity, the authors of Ref. (6) did not conclude damaging structural failure of catastrophic nature.

3.30 Derivation of Flow Quantities

Basin flow requirements as a function of time after shutdown are given in Figures 1, 2 and 3, based on maintaining the bulk basin coolant discharge temperatures below 80 degrees C to minimize damage to the basin structures as discussed above. The basic data and assumptions used to determine the flow requirements are discussed in Sections 3.10 and 3.20.

The required basin flow rates were obtained by assuming sensible heating of the basin coolant as indicated by the following equation:

$$T = T_i + \frac{Q}{WC_p} \quad (1)$$

T is the bulk coolant discharge temperature, T_i is the inlet coolant temperature, Q is the heat generation rate, W is the mass flow rate (mass/time), and C_p is the specific heat.

The second term on the right side of the equation is a measure of the temperature rise of the fluid caused by the heat input.

In determining the flow quantity (W), a differential temperature gradient of 68 degrees C was assumed for winter operation, and 56 degrees C for summer operation. Since high tank temperatures generally run about four degrees C above normal water inlet coolant temperature, bulk coolant discharge temperatures using the flow requirements in Figures 1, 2 and 3, and pre-shutdown power levels given in Table I, are calculated to be 77 degrees C in the winter and 80 degrees C in the summer.

The heat generation rate (Q) following discharge of the fuel elements from the reactor originates from the decay of fission products and was assumed to vary with time according to the predictions of Untermeyer and Weills, as reported in reference (7). The total heat generation rate is the sum of the heat generation of the stored metal and the heat generation of the newly discharged metal. The heat generation rate of the stored metal was determined using data obtained from Production Scheduling shown in Table II above.

(6) D. L. Condotta and H. G. Johnson to W. M. Harty, "A Review of Alternate Methods of Handling Effluent From a Primary Load Dump," dated October 8, 1962.

(7) Jones, S. S., "Heat Generation and Total Heat Output From the Pile After Shutdown," HW-33870, 11/23/54, Secret.

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During the discharge period sufficient flow is available so that the basin differential temperature increase should be less than 10 degrees C. Therefore, with an 80 degree C basin coolant discharge temperature limit, advantage may be taken of the available heat storage capacity of the coolant so that initial flow rates may be less than those predicted by equation (1). By so doing, limited coolant supplies may be conserved. The initial flow rates were determined as follows:

The amount of heat stored is the difference between the heat input and the heat removed; or

$$Q_{in} - Q_{transferred} = Q_{stored}$$

The rate of heat storage may be represented by

$$C_p W_T \frac{d(\Delta t)}{d-\theta} = Q_{in} - W C_p \Delta t \quad (2)$$

Where W_T is the weight of the water in the basin and $\frac{d(\Delta t)}{d-\theta}$ is the increase in the coolant differential temperature with time. Solving the equation for θ yields the following expression:

$$\theta = \frac{\log e \left[1 - \Delta t (K_1/K_2) \right]}{K_1} \quad (3)$$

where: θ = time to reach a given temperature increase (Δt)

$$K_1 = W/W_T$$

$$K_2 = Q/W_T C_p$$

Using this equation the initial flow rates which satisfy the basis were obtained by trial and error.

Referring to Figures 1, 2 and 3, the bulk basin discharge coolant temperature at the time of evacuation is assumed to be 10 degrees C above inlet coolant temperature. The initial flow rate is determined so that the basin discharge coolant temperature will gradually increase to a value near the 80 degrees C limit. If the flow rate were constant at this initial value the basin discharge coolant temperature would begin to decrease after 1-1/2 to 2 days because of the decrease in the heat generation rate. Following the solid curve, flow rates are sufficient to maintain the basin discharge coolant temperature below 80 degrees C. If flow quantities are not limited, it would be advantageous to follow the dashed portion of the curve during initial stages to provide added assurance that basin structural damage does not occur.

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3.31 Local Heating and Experience Verification

The flow requirements shown in Figures 1, 2 and 3, are not expected to prevent local boiling in a pile of discharged fuel elements. An experiment was performed at D reactor in which about one ton of freshly discharged enriched metal was stacked in a close packed array on a metal plate containing thermocouples. The temperature at the bottom center of the stack reached an equilibrium temperature which was 12 degrees C above the temperature of the coolant surrounding the stack. These data show that boiling could occur within a random pile containing several tons of fuel elements. Although the fuel elements will be adequately cooled (see discussion below), local coolant temperatures on the concrete floor beneath the fuel element pile may exceed the 80 degrees C. If boiling occurs the temperature could reach 110 degrees C. The lowest local fuel and basin temperature would be obtained if all metal were contained in buckets.

3.32 Circulation and Stratification

In the case of cooling fuel elements in the basin the bulk coolant velocity will be low, and the transfer of heat will be governed by the effect of temperature differences causing natural convection currents. If the coolant is introduced through the lower reactor process tubes, flow will pass under the wall separating the basin from the reactor rear face. This will direct the flow towards the bottom of the basin and the fuel elements and promote circulation and mixing of the coolant in the basin. At some reactors, notably DR, the majority of the coolant will be introduced on the basin side of this discharge wall. If this coolant is spilled into the basin from the discharge end of a pipe, there is concern that the coolant may stratify and be discharged without effectively cooling the fuel elements or the basin structure. Circulation and mixing can be improved by discharging several feet below the surface of the water, and directing the flow back toward the rear face of the reactor away from the basin discharge side. Such provision should be made in the design of these piping systems.

It is important to point out the undesirability of leaving any significant amount of metal in the chute. Such a pile would, to a great extent, result in a volume of metal which was inadequately cooled even though all of the fresh cooling water is introduced at the rear. This might occur because the hot water would remain trapped in the upper portion of stratification since the only exit for the water addition is downward by density current flow. See Figure 4.

This effect would be most pronounced at 105 B-D-F and perhaps the Ks, and would be of concern primarily to avoid a potential release of steam or fission products to the rear face which would cause delay in starting up again, either because of contamination or vapor damage.

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While final plans remain for the future, the authors suggest that depending on the nature of the alert, every effort be made to remove substantially all metal from the rear face chute area prior to actual evacuation.

3.40 Miscellaneous Considerations

3.41 Shield Temperatures

Following evacuation of the plants, continued coolant supply to the shields cannot be assured. The effect of loss of shield cooling after shutdown was investigated at the KW reactor during January 1963, by shutting off the shield coolant supplies shortly after shutdown and periodically recording shield temperatures as indicated by existing shield thermocouples. The following data were obtained:

<u>Time after Flow Stopped-Hrs.</u>	<u>Shield Temperature - C</u>		
	<u>Bottom</u>	<u>Top</u>	<u>Side</u>
2	23	10	10
24	27	9	9
72	25	7	6
With resumption of flow at $t \approx 72$ hours	12	6	5

As shown by the data, there was sufficient heat loss without coolant flow to prevent an appreciable shield temperature increase.

3.42 Reactor Tubes

With the fuel elements completely discharged, heat will be generated within the reactor due to the decay of radioisotopes in the process tubes, graphite, process tube surface film, and the shields. Since the reactor block will not be water-cooled during the evacuation period, the course of action proposed will meet the ground rules only if the process tubes maintain sufficient integrity to permit the resumption of reactor operations without costly tube replacement following the emergency. In a K reactor with Zircaloy-2 process tubes, the maximum process tube temperatures to assure that the tubes are not damaged is 250 degrees C. Above this temperature severe hydriding of the Zircaloy-2 tubes could occur. (8)

The aluminum process tubes are not subject to reactions with air atmospheres in the presence of water vapor up to temperatures of about 350 C. According to the Corrosion Handbook, at temperatures above 350 degrees C, internal oxidation of the aluminum could occur.

Water will be trapped in the tubes due to the configuration of the channels caused by graphite distortion. If the water remains in the liquid phase, mild pitting of the aluminum may occur over the two week

(8) Personal communication - R. L. Dillon

period. If the heat generation rate is sufficient, the water may vaporize and expose the tubes to atmospheric steam. Steam causes a definite protective film on aluminum alloys which is highly protective up to 180 degrees C to 200 degrees C. At temperatures above this range the steam reacts with aluminum with the formation of aluminum oxide and hydrogen under some conditions. Further, intergranular corrosion of the 1100 aluminum alloy occurs at coolant temperatures of about 200 degrees C.

It is the conclusion of the authors that the process tubes will not be significantly damaged if the stack temperature rise is limited to 200 degrees C in either the aluminum or Zircaloy-2 tubed reactors. This would permit steady state heat generation rates for the two week period of about 200 kw in the six older reactors and 300 kw in the K reactors. Preliminary calculations indicate that the heat generation rates will be less than one-half these values. The Radiological Engineering Unit, R&EO, is currently performing calculations to refine the preliminary estimates.

3.43 Fuel Corrosion

Fuel element cladding temperatures could reach 110-120 degrees C during the emergency in sections of the fuel element stack where local coolant boiling is likely to occur. The corrosion rate of the aluminum cladding will be less than it is under reactor operations, or less than about 2 mils/week. Thus, it is not expected that fuel element jackets will corrode away and expose a large uranium surface to corrosion.

Local pitting attack may be experienced. Pitting to the uranium could expose a small uranium surface to the coolant. The uranium water reaction at these temperatures is slow, and undercutting of the aluminum to uranium bond should not occur. Should the bond be destroyed, the corrosion products should be largely contained within the jacket. Therefore, contamination of the basin from the effects of corrosion is expected to be minor.

3.44 Outcome of Coolant Loss to Basins

If the plants cannot be re-entered within three to five weeks following the alert to re-establish basin coolant supplies, the emergency coolant will be exhausted and the basin water will eventually be lost by the combined effects of boiling and leakage. Assuming that coolant flow is available for three weeks following evacuation of the plants and that the rates of coolant loss by boiling and leakage following the depletion of coolant supplies are the same, it would require about ten days for a basin to boil dry. During this period the fuel elements would be adequately cooled.

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Following loss of water, heat transfer would occur due to natural convection and radiation in the air environment. Calculations by Zaloudek⁽⁹⁾ show that the fuel element temperature will be well below the cladding melting temperature if the fuel element surface is completely exposed to the natural cooling effects of air. However, those fuel elements located in the central regions of storage buckets or in stacks on the basin floor would reach higher temperatures because of poorer heat transfer conditions. Using a packed bed analogy and neglecting the effects of natural convection, it is calculated that the cladding of some of the fuel elements in the fuel element stacks could melt. The uranium of these fuel elements would not be expected to melt but would be subject to oxidation and may burn. It is concluded that the eventual loss of water to the basins could result in melting of the cladding of a small percentage of the fuel elements, and, by oxidation of the exposed uranium in partial release of the fission products in these fuel elements.

In addition, the basins have accumulated considerable radioactive sediment - principally from fuel elements which failed during reactor operation and were discharged into the basins. The airborne dispersion of this sediment, if the basins were to dry out, is also possible.

No criticality incident would accompany the event because none of the normally used Hanford fuel elements can form a critical array without water even if they are physically rearranged.

3.45 Control Requirements

The discharge will be performed within process standards requirements such that a critical arrangement of the enriched fuel elements will be prevented by physical means. No further control requirements are necessary to protect against a criticality incident.

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Evacuation @ 2-2/3 days after shutdown
Maximum basin coolant discharge temperature - 80° C

Evacuation @ 2-2/3 days after shutdown
Maximum basin coolant discharge temperature - 80° C

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Evacuation at 3-2/3 days after shutdown
Maximum basin coolant discharge temperature - 80° C

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Fig. 4

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B-D-F
Reactors

Potential Boiling
From Stacked Metal
in Rear Face Chutes

Shield
Wall

→ Probable Water Circulation

Potential
Boiling

Rear Face of
Reactor

Plank Deck

Water Level

Cold
water

Void
Space

DATE

FILMED

10/18/94

END