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PRELIMINARY HAZARDS REVIEW
OVERBORING HANFORD REACTORS

R. Nilson and P. A. Carlson

INTRODUCTION

The General Electric Company, as prime contractor to the AEC at Hanford, is proposing to modify the lattice characteristics of the 8 3/8-inch lattice reactors for the purposes of improving the conversion ratio of these reactors. The proposed overbore modification of the reactors would remove the existing aluminum process tubes, enlarge the diameters of the graphite channels by about one-half inch, insert smooth-bore Zircaloy-2 process tubes and refuel the reactor with larger size, self-supported fuel elements. The overbore fuel will remain the internally-and-externally-cooled cylindrical type, but the weight per foot will be about twice that of the present fuel element.

The removal of the inlet and outlet piping connections which would be required in the overboring process will permit the replacement of the existing fittings with ones of improved design. Furthermore, new orifices and venturis which are compatible with the hydraulic characteristics of the overbore tube and fuel geometry and the pumping system will be installed. No basic changes are proposed in the pumping system* though the reactor flow rate may be increased 5 - 10 per cent by changes in hydraulic characteristics depending on the water plant flow capacity.

*The overbore proposal is part of an over-all plant improvement program which also includes providing automatic, fail-safe, last-ditch coolant systems for all reactors. However, this latter item will also be requested even if overboring is not approved. (The present export system now used for last-ditch cooling does not meet reliability criteria.)

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The modification will double the fuel-to-graphite ratio of the reactors resulting in an undermoderated lattice. A nuclear chain reaction will not be possible with natural uranium fuel and the use of some slightly enriched fuel will be required to sustain the reaction throughout the entire operating cycle. The necessary enrichment level will probably be obtained by the use of a two-zone loading similar to that used at present to flatten the radial flux distribution.

Throughout the text the pertinent features of the overbored reactor will be compared to the present reactor. In this way, a relative assessment of the nuclear safety is presented. All calculations have been made for C Reactor, but the conclusions apply equally to all six older Hanford reactors.

SUMMARY AND CONCLUSIONS

It is concluded that the over-all nuclear safety of the 8 3/8-inch lattice reactors will be materially improved by the overbore modification. This conclusion is based on the following changes in the basic physics and engineering features of the reactor:

- 1) The overbore lattice will be "fail-safe" upon loss of coolant. Instead of gaining up to 3.5% in reactivity upon total loss of coolant (as is the case in the present C lattice and fuel geometry), the overbore reactor will lose reactivity (one to two per cent depending on conditions).
- 2) The prompt, negative, fuel temperature coefficient of reactivity in the overbore reactor will be nearly double that in the reactor at present.
- 3) The positive temperature coefficient of reactivity associated with heating the moderator is smaller in the overbore reactor by at least 20 per cent.
- 4) The maximum excess reactivity (non-accident) is less in the overbore reactor than in the present reactor.
- 5) The overbored lattice will gain reactivity in the event the core is flooded. However, the maximum conceivable reactivity gain due to flooding would occur in the hot core and would be less than one per cent. In the flooding calculation, the entire reactor is assumed flooded, no water is assumed flushed to steam and all the water is assumed to collect in the position of maximum reactivity effect. This maximum calculation thus describes a physically impossible event.
- 6) The net reactivity effect associated with increasing core temperatures to the melting point of uranium is considerably more negative in the overbore reactor than in the present reactor.
- 7) The effective nuclear strengths of the control systems will be reduced in the overbore reactor. In spite of the reductions, either vertical control

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system will individually meet accident control requirements. This is not the case in the present reactor; under certain conditions, only the Ball-3X System is strong enough to hold the dry reactor subcritical.

- 8) The reduction in the strength of the horizontal control rods will result in increased use of the poison splines now utilized to supplement the horizontal rods.
- 9) The potential for fuel melting following a flow-loss or power-surge transient will be no greater in the overbore reactor than in the present reactor and will probably be less due to the greater heat capacity of the fuel elements.
- 10) Assuming the same equilibrium power levels, the emergency coolant flow rate required to maintain satisfactory cooling in the event of a loss of primary and secondary pumping power will be less in the overbore reactor during the early period when flywheel decay and high tank flow are the major sources of water. During the later period (greater than an hour following power loss) when the export system flow alone is available, the flow demand for the overbored reactor will be slightly greater than that for the present reactor.

The types of accidents to which a Hanford production reactor is potentially subject are not significantly changed by the overbore project though the changes outlined above will affect the probabilities and severities. A complete meltdown of the core would release fission products to the reactor building and eventually to the environs. The inventory of fission products will in general be increased by the modification of the overbore project. A useful comparison with a Hanford K Reactor can be made. The fuel inventory in an old reactor is currently about half that of a K Reactor and will be increased to approximately equal the K inventory. If the thermal power level of the reactor is unity, that of the K Reactor is about two and the overbore project will increase the reactor power to at most 1.2. Over-all then, the long-lived fission product inventory is expected to approximate that in a K Reactor today, while the short-lived products, being primarily power level dependent, will remain substantially below the level currently carried in a K Reactor. The consequences of a meltdown accident in an overbored reactor are thus expected to be somewhat less severe than those from a similar accident at current K-Reactor operating conditions.

In total, the modifications to the old reactors under the proposed overbore modification improve the safety status of these reactors. The increase in fission product inventory, though undesirable in itself, is more than balanced by the improved lattice characteristics. It therefore is judged that the nuclear hazards associated with reactor operation will not be increased, but will in fact be reduced by the proposed modification.

SITE INFORMATION

Since the proposed project involves modifications to existing Hanford production reactors, the usual information such as site location, description, etc. is not given in this preliminary review.

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REACTOR*

C Reactor is a graphite-moderated, light water-cooled, heterogeneous, thermal production reactor. The maximum presently-authorized power level is 2310 MW, corresponding to a power density of 3.5 watts/cm³ over the active volume of the pile. The maximum level reached to date is 2175 MW, which occurred during the 1960-61 winter. The maximum power level assumed for C Reactor in this review following the proposed alteration is 2310 MW. The reactor has operated since 1953.

The graphite which serves as the moderator in the reactor is also used for the structural material of the reactor. The design bases for the graphite in C Reactor, in addition to nuclear purity and compressive strength, required a layup assembly designed to withstand a Zone II earthquake, as defined by the Uniform Code of the P.C.B.O.C.** , 1950.

Heat generated by the nuclear process is removed from the reactors by passing light water through the nearly 2000 individual, fuel-bearing process tubes. The coolant is not recirculated. The tubes are horizontal and are arranged in an 8-3/8 inch square lattice. The present process tubes are 28 aluminum and projections, called ribs, extend the length of the tubes to support the fuel elements. The proposed overbore design will probably call for ribless, Zircaloy-2 process tube. Fuel support would then be accomplished by supports attached to the fuel elements, the so-called self-supported element.

The primary pumping system which delivers treated river water to the reactor is electrically driven and is backed up by a steam-driven system. Auxiliary sources of coolant are also available and these include high tanks and an inter-area export water system.

The fuel in the Hanford reactors is predominantly natural uranium. A small amount of slightly enriched uranium fuel (0.947 w/o U235) is used to provide excess reactivity and power flattening. The fuel geometry is tubular, enabling coolant to flow within the fuel element as well as outside of it. The cladding is X-8001 aluminum alloy bonded to the uranium by a layer of AlSi. (The tubular fuel design has been in service for approximately five years, replacing the original solid Hanford fuel element). The overbore fuel elements will be similar except for size and the fact that fuel support in the process tubes will be accomplished by supports attached to the fuel element. The expected weight per foot of the overbore fuel element is twice that of the present fuel element. To support the loss in reactivity incurred by increasing the fuel size, additional amounts of enriched uranium will be required in the modified reactor. Refueling the reactor is done manually, a tube at a time.

Operating, startup and normal shutdown transients in C Reactor are controlled with 15 control rods. The rods enter from one side of the reactor perpendicular to the axes of the fuel channels. The control rods are supplemented by poison strips called splines which are inserted in the fuel-bearing tubes to meet control requirements under certain circumstances. The use of such supplemental poisons is now a standard practice at Hanford.

Emergency shutdown control is effected by dropping boron-steel safety rods or small boron-steel balls into vertical reactor channels. At C Reactor there are 45 such vertical control channels.

* All subsequent discussion is in terms of C Reactor. However, the statements and conclusions for the five older old reactors (B,D,F,DR and H) differ only in degree.

** Pacific Coast Building Officials Conference held in 1950.

The fundamental, nuclear and physical, process variables such as neutron flux, coolant temperature and flow, and graphite temperature are continuously monitored by instrumentation. Reactor shutdown (in many instances automatic) is called for if these critical process variables exceed limits or upon earthquake and loss of power.

The proposed modification will not alter the hardware of the control and instrumentation systems. However, the nuclear properties of the lattice will be sufficiently changed to alter the strengths of the various control poisons. These changes are covered in the Control Systems Section later in this report.

The Hanford reactors are not enclosed in containment structures. A reactor confinement program which affects all the eight existing production reactors, provides fog sprays in the rear face work areas, filters to remove particulate matter from the exhaust air from the reactor buildings and modifications in the reactor building ventilation system. No additional confinement is planned in the proposed modification.

Additional reactor data are given in Table I.

TABLE I

FUEL AND PROCESS TUBE INFORMATION

	<u>C-Present</u>	<u>C-Overbore</u>
Fuel Geometry	I&E ^a	I&E ^a
Fuel Cladding	X-8001 Alum. Alloy	X-8001 Alum. Alloy
Fuel Dimensions (inches)		
Uranium ID	0.481	0.452
Uranium OD	1.370	1.883
Clad ID	0.375	0.339
Clad OD	1.460	1.991
Fuel Weight (lbs/ft)	9.97	20.25
Fuel Support	Process Tube Ribs	Self-Support
Tons Fuel (1980 Tubes)	220	470
Average Enrichment (w/o U ²³⁵) ^b	0.75	0.81
Specific Power (kw/kg) ^c	10.7	5.42
Goal Exposure (MWD/T)	800 ^d	800 ^d
Total Energy Extracted Per Unit Fuel		
Volume (MWD/cm ³)	0.0167	0.0167
Process Tube	2S Aluminum	Zircaloy-2
Process Tube ID (inches)	1.633	2.147
Process Tube OD (inches)	1.764	2.250

^aInternally-and-externally cooled

^bPower-weighted average.

^cAt 2175 MW for C-Present and 2310 MW for C-overbore.

^dAssumed in this report. Present exposure levels are lower.

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HYDRAULICS AND HEAT TRANSFERPressure Drop Characteristics

A comparison of the hydraulic characteristics of the present and overbore equipment is shown in Table II. The greatest hydraulic difference between the two systems is the pressure drop in the process tube inlet fittings. This pressure drop in the overbore case is less than one-half that in present, C-Reactor inlet fittings. This decreased pressure drop will result in an increase in reactor flow rate. (Alternately, the fuel element could be designed to absorb the difference.) For the reactor completely charged with the overbore fuel of the size specified in Table I, the reactor flow will be about six per cent higher than now and the pressure drop across the reactor will decrease about ten per cent. (Ten pumps and the present orificing efficiency are assumed).

TABLE IIPressure Drop Characteristics of C-Reactor Fuel Geometries

	<u>Present</u>	<u>Overbore</u>
Reactor Flow Rate, gpm	100,700 ^a	107,000 ^b
Tube Flow Rate in Central Zone, gpm	51.8	55
ΔP , Front Header to Active Zone, psi	125	56
ΔP , Across Active Zone, psi	217	247
ΔP , Active Zone to Rear Header, psi	20	25
ΔP , Front Header to Rear Header, psi	362	328

^aPresent C-Reactor flow rate with 86 per cent orificing efficiency and ten pump operation.

^bEstimated reactor flow rate with the reactor completely loaded with the overbore geometry; orificing efficiency same as present case and ten pumps.

Heat Transfer Characteristics (Normal Operation)

The heat transfer characteristics of the present and overbore geometry are compared in Table III at the expected reactor operating conditions. The uranium core temperature in both systems is well below the limiting alpha-to-beta phase transformation temperature of 660 C. Aluminum surface temperatures and corrosion rates are expected to be lower in the overbore system because of lower surface heat fluxes and improved centering of the fuel element in the process tube. For example, in the present ribbed tubes the fuel elements operate with a thinner cooling annulus at the top of the tube than elsewhere during the period before significant tube corrosion has occurred. This results in water temperatures which are about 25 per cent higher at the top of the annulus in new process tubes. As the tubes corrode in service and the rib height decreases, this value is gradually reduced until the annulus between the ribs is limiting. Thus, it has been found economical on a tube-life basis to purposefully design for higher, initial top-of-annulus temperatures.

The self-supported, overbore geometry is slightly eccentric toward the bottom of the tube because of the need for charging tolerances on support design. In this case, the eccentricity results in only about ten per cent higher water temperatures at the bottom of the annulus.

The temperatures shown in the table for the present fuel geometry are the values for new, ribbed process tubes. The temperatures shown for the overbore system are for the case of smooth-bore, zirconium tubes and self-supported fuel elements. In the latter case, the temperatures would be expected to remain at nearly the same level throughout the life of the tube for given flow, power and film conditions.

TABLE III

Heat Transfer Characteristics Present Equipment vs. Overbore Equipment

	<u>Present Equipment</u>	<u>Overbore Equipment</u>
Reactor Flow Rate, gpm	100,700	107,000
Central Tube Flow Rate, gpm	51.8	55
Reactor Power, MW	2175	2310
Maximum Tube Power, kw	1360	1446
Maximum Specific Power, kw/ft of charge	77.6	82.5
Maximum Heat Flux, Btu/hr-ft ²		
Inner Surface of Fuel	867,200	820,700
Outer Surface of Fuel	475,000	403,400
Burn-Out Heat Flux, Btu/hr ft ²		
Inner Surface of Fuel	3.4 x 10 ⁶	4 x 10 ⁶
Outer Surface of Fuel	2.2 x 10 ⁶	2.4 x 10 ⁶
Inlet Water Temperature, °C	13	13
Reactor Bulk Outlet Water Temperature, °C	95	95
Maximum Tube Bulk Outlet Temperature, °C	113	113
Maximum Uranium Core Temperature, °C	315	365
Maximum Aluminum Surface Temperature, °C		
Inner Surface of Fuel	123	119
Outer Surface of Fuel	145	130
Corrosion Rate, mils/month		
Inner Surface of Fuel	6.3	5.0
Outer Surface of Fuel	9.0	3.6
Coolant Velocity, ft/sec: Inner Coolant Passage	47	50
Outer Coolant Passage	28	26
Elastic Stress, psi: Inner Surface of Fuel	57,400	76,400
Outer Surface of Fuel	61,600	80,200

The elastic stress in the fuel element surface is about 30 per cent higher for the overbore case. This increases the potential for uranium core failures. However, since uranium does not behave in an elastic manner, these values are only indications that the core failure rates will be more severe. The solid fuel elements, which preceded the I&S fuel elements at Hanford, suffered core failures at a calculated elastic stress of 110,000 psi which is about 38 per cent greater than that calculated for the overbore fuel element. Because of differences in geometries and associated lower temperatures, a direct comparison is not valid. The uranium core behavior must therefore be evaluated in the reactor.

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The burn-out heat fluxes at normal operating conditions are also shown in Table III. The values of burn-out heat flux were calculated using the correlation of Mirshak, Durant, and Towell,¹ which has been found to correlate a limited number of experimental points obtained with Hanford geometries in the Hanford Laboratories.² The calculated burn-out heat fluxes were divided by two to obtain a conservative operating value. These values therefore represent a conservative value of the maximum heat flux required for film boiling and burn-out.

Because of the varying heat distribution along the process tubes, the difference between the burn-out flux and operating flux varies from point to point along the length of the channel. For a cosine distribution, the closest approach to the burn-out flux occurs at a point about seven feet from the downstream end of the fuel element charge. Since the burn-out would occur over a short length in the process tube, the consequences of such an event would be limited to those associated with a fuel-cladding failure.

As shown in the table, the burn-out heat flux exceeds the operating values by three to four times for both geometries. The peak operating heat fluxes were determined by applying factors to account for front-to-rear peaking (15%), the effect of end-caps (12%) and the effect of supporting ribs (10%), but no fuel misalignment was assumed. Burn-out tests made at Hanford to investigate the effect of fuel-tube eccentricity on burn-out³ show that the burn-out flux is markedly reduced by fuel element misalignment. For example, if the fuel element is raised off the ribs so that 90 per cent of the original top annulus thickness is displaced by the fuel element, the burn-out flux is on the order of 500,000 Btu/hr ft² because of the altered coolant conditions. The self-supported fuel element concept prevents misalignment of the fuel element in the process tube. Thus, these fuel elements are not subject to the "hot spot" failures experienced with present geometries. This has also been verified by in-reactor testing.^{4,5}

Transient Heat-Transfer Characteristics

As shown in the preceding section, the heat-transfer conditions are satisfactory during normal reactor operation with either the present or overbore fuel geometries. There is no boiling in the process tube and heat transfer rates are high enough to maintain fuel element surface temperatures near bulk water temperatures. A loss of flow or a power level surge will alter these steady state conditions. At present, in C Reactor, a reduction of about 30 per cent in the flow rate in the highest power tubes causes flow instability resulting in a rapid transition to heat transfer conditions under which the fuel elements would overheat and melt if the reactor were not

1. DP-355, Heat Flux at Burn-Out, S. Mirshak, et al, February 1959 (Unclassified)
2. HW-65438, Sub-cooled Burn-Out Relative to Hanford Reactors, P. A. Carlson and R. E. Trumble, May 31, 1960 (Secret).
3. HW-71024, Experimental Boiling Burn-Out Conditions for Hanford Production Reactors, E. D. Waters and J. M. Batch, September 20, 1961 (Secret).
4. HW-59756E, Irradiation Summary Report, PT-IP-247A-FP, Irradiation of 1.47 Enriched, Self-Supported, I&E Fuel Elements in Ribless Process Tubes, R. E. Hall, April 7, 1960 (Secret).
5. HW-65717, Status Report: Project Fuel Testing Program, M. Clinton and D. W. Peacock, July 15, 1960 (Secret)

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shut down. It is necessary, therefore, to continuously monitor the flow condition in each process tube.

Flow-monitoring is accomplished in the Hanford reactors with pressure-sensing gauges connected to the low pressure tap of a flow-measuring device (orifice or venturi). These orifices or venturis are located between the front crossheader and the inlet to each process tube. During normal operation, the comparison of the pressures at these points to the front crossheader pressure gives an accurate measurement of the tubes' flow rates. By high and low pressure trips, incorporated electrically into the reactor safety system, the gauges automatically initiate a reactor scram under conditions of flow inadequacy.

The trip settings of the pressure monitors and the tube power limits are determined by the response characteristics of the pressure-sensing gauges and the transient heat-transfer behavior of the system. These characteristics are determined by simulating flow loss and power surge accidents in an electrical-resistance, heater-tube laboratory mockup of the reactor process tube.

The effect of overboring at C Reactor on burn-out during a flow-loss accident is shown in Figure 1⁶. The plot is of flow versus tube power. The hatched area on the graph labeled "operating region" shows the possible range of normal operation with the maximum operational tube outlet temperature permitted being 130 C. The reactor is automatically scrambled by high or low trips whenever the flow rate falls to a value which would result in a 160 C outlet temperature at the peak of the transient. A temperature of 160 C is about the point at which flow instability would be initiated in the process tube.

The region in Figure 1 between the trip line CD and the film boiling lines AB and A'B' is a region of two-phase metastable flow, where reductions in flow cause the pressure drop to increase and further reduce the flow. Operation can be stable at points to the left of the film boiling lines AB and A'B'; however, flow rates below these values will result in surface temperatures higher than the melting point of the aluminum cladding and burn-out will occur.

Experimental data which have been obtained in the heat transfer mockups show that the flow rate at which film boiling is reached in the process tube for a given power level is about 20 per cent higher for the overbore fuel elements. The reason for this increase is not yet fully understood. Contributing factors are believed to be (1) differences in the two-phase flow characteristics of the outlet fittings and fuel element charges, and (2) differences in the flow area which results in lower velocities at a given flow rate for the overbore system.

The behavior of the highest power tubes for the reactor operating conditions listed in Table III are shown by the line a, b, c, d for the overbore case and line e, f, g, h for the present reactor case. For the overbore case, the normal operating parameters for the highest power tube (flow 55 gpm, tube power 1446 KW, and outlet water temperature 112 C) define the point a. The minimum permissible operating flow at this power level is seen to be 46 gpm which produces a 130 C tube outlet water temperature (point b). Operation at flow rates between 46 and 37 gpm is in the stable flow region, but requires reactor shutdown. The reactor is automatically scrambled by the pressure

6. HW-71075, Results of Laboratory Heat Transfer Experiments for C Reactor Overbore, Fuel Channels, E. D. Waters and H. S. Krieter, November 10, 1961.

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FIGURE 1

FLOW VS. POWER LEVEL

Tube Flow - gpm

instrumentation if the flow rate is reduced below 37 gpm (point c). At flow rates less than 37 gpm the flow is unstable and rapidly approaches point d where the downstream fuel elements would operate with film boiling of the coolant. The flow continues to decrease past point d into the burn-out region if the reactor remains at full power. Similarly, the points, e, f, g and h represent the present reactor case.

Experimental data have shown that the initiation of the reactor scram can be delayed several seconds beyond point d and still fuel element melting will be avoided. Thus, flow loss beyond point d does not cause burn-out if the reactor is scrambled promptly.

As an example, the worst accident studied in the laboratory tests on the overbore geometry was for an instantaneous flow loss of 70 per cent at a tube power of 1800 kw. In this experiment, the initial flow rate was 65 gpm. The flow rate, immediately following the simulated process tube plugging, was 19.8 gpm. As shown by Figure 1, film boiling is initiated for this power level at 28.4 gpm. Because of flow instability, the flow continued to decrease beyond 19.8 gpm. At a flow rate of 8 gpm, the experiment was terminated when the rod temperatures indicated a temperature corresponding to the aluminum melting temperature. This occurred about ten seconds following the plugging at 19.8 gpm. In the reactor, the scram would have been effective in less than three seconds and fuel element melting would not have occurred.

For the overbore size fuel elements, the time to reach the aluminum melting temperature following the beginning of film boiling in the process tube is longer by at least a factor of two. This is due to the greater heat capacity per unit length of the larger overbore fuel elements, which results in a slower rate of temperature rise following a transition into the burn-out flow region. Thus, although film boiling begins earlier in the transient for the overbore fuel elements, the aluminum melting temperature or burn-out is reached a few seconds later than for present size fuel elements. It may be concluded on the basis of these data that the potential for fuel element melting during flow-loss or power-surge transients is no worse and may be reduced by the use of the overbore fuel elements.

PHYSICS

Reactor Loading Pattern

The Hanford reactors are operated with an artificially flattened flux distribution in the side-to-side and top-to-bottom directions. (The front-to-rear distribution follows the natural cosine shape.) The flattening is accomplished by the use of slightly enriched fuel elements (0.947% U²³⁵) in the lattice units near the periphery of the pile.

The overbore reactor, if uniformly enriched, would require an enrichment level in the neighborhood of 0.81% U²³⁵. The necessary enrichment level can be accomplished by extending the present two-zone loading concept so that the requisite flattening can also be attained. The following loading arrangement has been chosen for the purposes of this study. A central zone, roughly cylindrical, is made up of 69 per cent natural uranium fuel columns and 31 per cent of 0.947% enriched uranium fuel columns. Surrounding the central zone is an outer zone made up of 45 per cent natural and 55 per cent enriched fuel, respectively. The loadings for C Reactor at present and for the assumed overbore pattern are shown in Figures 2 and 3.

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FIGURE 2
NORMAL LOADING PATTERN FOR C REACTOR

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FIGURE 3
ASSUMED TWO-REGION LOADING FOR C OVERCORE REACTOR

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Temperature Coefficients

The Hanford reactors experience two temperature-dependent reactivity effects. Heating the uranium produces a negative reactivity effect, whereas graphite heating causes a positive effect. Density changes in material cause only minor variations in reactivity. The following discussion is limited to temperature effects in the wet reactor, that is, with coolant in the channels. The dry-pile coefficients are significantly different and are discussed later.

Doppler Temperature Coefficient of Reactivity

In predominantly natural uranium or low-enriched uranium fuel reactors, the Doppler temperature coefficient of reactivity is negative. Because of its prompt nature, the Doppler effect exerts an inherent temperature-stabilizing effect in the reactor. The Doppler coefficient for C Reactor with the present fuel loading is $1.9 \times 10^{-5}/\text{C}$. This is an averaged value over an uranium temperature range of 20 to 230 C. Recent measurements^{7,8} have indicated that the Doppler coefficient is not a linear function of temperature. For C Reactor, the Doppler coefficient can be expressed as

$$\frac{1}{k} \frac{dk}{dT} \approx 3.5 \times 10^{-4} (T + 273)^{-1/2}/\text{C}.$$

The Doppler coefficient for the overbored C Reactor is expected to be nearly twice that for the present reactor due to the nearly double importance of resonance capture on the neutron balance in the new lattice. The Doppler coefficient in the C overbore reactor will be

$$\frac{1}{k} \frac{dk}{dT} \approx 6.5 \times 10^{-4} (T + 273)^{-1/2}/\text{C}.$$

The total Doppler defects in the operating reactor are given in Table V.

Graphite Temperature Coefficient of Reactivity

The graphite temperature coefficient is positive in the Hanford reactors. The coefficient is smallest for green fuel; the coefficient increases with exposure chiefly due to the influence of plutonium-239.

The effect of overboring is expected to be a reduction of about 20 per cent in the graphite coefficient. This is due to a combination of effects. The smaller effect of temperature on the thermal utilization (due to the increased fuel volume) accounts for most of the reduction. The effect of the higher enrichment on the graphite temperature coefficient in the overbore reactor is small due to nearly compensating effects on the coefficients $1/f \, df/dT$ and $1/\eta \, d\eta/T$.

7. Brimberg, S. and Dahlstrom, J. (1958) Proceedings Second Int. Conf. on Peaceful Uses of Atomic Energy, Geneva, 1958 P/150, Vol. 16, 172. United Nations, N.Y.
8. HW-69343, The Variation of k_{eff} With Fuel Temperature for The NPR, R. I. Smith, April 21, 1961 (Confidential)

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The exposure-dependence of the graphite coefficient in the overbore reactor is not expected to differ greatly from that now obtained. The increased initial U²³⁵ concentration in the overbore fuel will reduce the positive effect of plutonium-239 buildup on the coefficient; but, on the other hand, a higher epithermal flux in the overbore reactor and a higher conversion ratio will tend to increase the importance of plutonium-239 on the reactivity balance.

Typical values of the graphite coefficient in the present C Reactor are shown in Table IV. These data are based on actual reactivity balances in the operating reactor. They should be accurate to + 20 per cent.⁹ The values for the overbored case are based on theoretical calculations, normalized to present reactor data.

TABLE IVGraphite Temperature Coefficient of Reactivity

<u>Average Pile Exposure</u>	<u>Present C Reactor</u>	<u>Overbore C Reactor</u>
0	$3.0 \times 10^{-5}/C$	$2.4 \times 10^{-5}/C$
200	$3.8 \times 10^{-5}/C$	$3.0 \times 10^{-5}/C$
400	$4.5 \times 10^{-5}/C$	$3.6 \times 10^{-5}/C$
600	$5.2 \times 10^{-5}/C$	$4.1 \times 10^{-5}/C$

The total graphite temperature reactivity contributions in the operating reactor are given in Table V.

Xenon Poisoning

The equilibrium xenon poisoning is flux-dependent. In C Reactor, at present, the equilibrium xenon reactivity effect is 2.6 per cent in the cold pile. The thermal flux in the overbore reactor is reduced by a little more than two since essentially the same power level will be achieved with over twice the U²³⁵ per unit reactor volume. The xenon poisoning is consequently reduced in the overbore reactor. The expected equilibrium value is 2.2 per cent.

Xenon-135 is unstable and its reactivity effect is not permanent. After reactor shutdown, the xenon poisoning initially increases due to the decay of the precursor iodine-135 and eventually decays to zero.

9. Apparent reactivity contributions due to graphite heating in the reactors do not match those which are theoretically estimated. Since the only graphite temperature coefficient data available are those based on pile experience, it has been judged most meaningful to accept these values, in spite of a rather large uncertainty. For comparison purposes, however, the error introduced should not be significant.

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Samarium Poisoning and Neptunium Holdup

Sm^{149} is a stable isotope and is the daughter of fission product Pm^{149} . Samarium poisoning reaches an equilibrium value at an exposure of $\sim 100 \text{ MWD/T}$ in the Hanford reactors. The poisoning is not flux-dependent and therefore should increase only slightly in the overbore reactor. The difference is due to a higher fission-to-absorption ratio in the fuel of the overbore reactor due to the higher effective enrichment.

The samarium poisoning is temperature dependent. Hot, equilibrium values are about $0.64\% \Delta k/k$ and $0.66\% \Delta k/k$, in the C present and overbore reactor, respectively. The poisoning effect in the cold reactor (from samarium produced in the hot, operating reactor) is about 30 per cent smaller. The observed graphite coefficients listed in Table IV should include the temperature-dependence of the samarium poisoning.

Unlike xenon, the samarium poisoning is a permanent effect due to the stable nature of the isotope. However, following a reactor shutdown, the samarium concentration increases due to the decay of its precursor. Following a startup, the samarium poisoning will regain its equilibrium value after about 100 MWD/T of additional exposure.

The calculation of exposure-dependent reactivity effects usually assumes instantaneous formation of plutonium. The fact that plutonium formation is delayed through the beta-decay of 2.33-day Np^{239} makes it necessary to correct calculated reactivity gains due to exposure. This can be done by assigning a reactivity "loss" for the neptunium holdup effect. Alternately one can use an observed graphite temperature coefficient in the reactivity balance which inherently includes the delayed effect. The coefficients in Table IV should include the neptunium holdup effect. One situation in which it might be important to consider the neptunium holdup is in the shutdown reactor. Fortunately, however, the reactivity loss due to increased samarium after shutdown very nearly balances the reactivity gain from the decay of neptunium into plutonium. For reference purposes, the neptunium holdup effect (the difference in reactivity between actual and instantaneous plutonium formation) is $0.4\% \Delta k/k$ and $0.2\% \Delta k/k$ in the C present and overbore reactor, respectively. The reason for the lower value in the overbore reactor is the fact that the thermal flux is lower by about 50 per cent and thus the equilibrium neptunium is reduced by the same amount.

Long-Term Gain (Cold)

The net reactivity effect of plutonium formation, U^{235} burnout and fission product formation (other than xenon and samarium) is defined as the long-term gain. Since the graphite temperature coefficient also includes these effects, it is logical to define the long term gains for the cold reactor.

Based on actual observation of long-term reactivity effects,¹⁰ the cold long term gain in natural uranium fuel attains a maximum value of $0.5\% \Delta k/k$ at an exposure of 550 MWD/T . Beyond this exposure, the buildup of higher plutonium isotopes reduces the gain.

Calculations indicate that the cold, long-term gains in the overbore reactor will be very nearly the same as in the present reactor.

¹⁰HW-46377, Long Term Reactivity Data from Irradiated Slugs - Final Report
PT-105-553A, C. E. Bowers and R. O. Gumprecht, February, 1957 (Secret).

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Reactivity Requirements (Detailed Breakdown)

The preceding discussion enables a complete breakdown of the reactivity requirements. Assuming 1% $\Delta k/k$ held in control rods in the equilibrium reactor, the approximate breakdowns are as follows:

Table VEstimated Reactivity Requirements

	<u>C-Present</u>	<u>C-Overbore</u>
Rods at Equilibrium	-1.0%	-1.0%
Doppler Coefficient	-0.5%	-1.0%
Xenon (Hot, equilibrium)	-2.6%	-2.2%
Samarium (Cold, equilibrium concentration)	-0.45%	-0.46%
Graphite Coefficient (250 MWD/T) ^a	+1.7%	+1.3%
Long Term Gains (Cold, 250 MWD/T) ^b	+0.45%	+0.42%
<hr/>		
Cold, clean, green ^c	2.4%	2.94%

^aIncludes samarium temperature coefficient, neptunium effect, temperature dependence associated with U²³⁵ burnout and plutonium formation and any temperature coefficient of control rods.

^bIncludes reactivity effect of U²³⁵ burnout, plutonium formation and fission product formation (other than Sm and Xe) in cold reactor.

^cCold, clean, green reactivity required to support chain-reaction in hot, operating reactor with average exposure of 250 MWD/T.

Maximum Excess Reactivity (Non-Accident)

The maximum excess reactivity which can be obtained during any time during normal reactor shutdown or operation depends not only on the magnitudes of the various reactivity effects but on their time-dependences.

To indicate the maximum excess reactivity, typical startup reactivity transients are shown in Figure 4 for the two cases. The curves in Figure 4 should represent the most extreme case insofar as excess reactivity is concerned during a startup. The curves are for a startup from an infinite outage (zero xenon). The average pile exposure is 600 MWD/T, which is the maximum normally attainable under present charge-discharge practices.

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FIGURE 4

STARTUP TRANSIENT

(600 MW/DT AVERAGE EXPOSURE)

C-REACTOR ∞-OUTAGE

REACTIVITY (PER CENT)

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The curves shown in Figure 4 are also based on the enrichment level required for one per cent excess reactivity at equilibrium at a 250 MW/D/T, average exposure. It is not unlikely that with the flexibility available from the expanded use of slightly-enriched uranium in the overbore pile the ratio of natural to enriched uranium will be changed during succeeding charge-discharge outages to help compensate for the reactivity variation with exposure.

As the curves indicate, the maximum reactivity for the overbore case is ~ 3.5 per cent as compared to the present case in which the maximum is ~ 3.8 per cent. The turnaround maxima are based on a 90 per cent rise to equilibrium power at turnaround. Slower rates of power ascension would increase the xenon inventory at turnaround and thus reduce the turnaround reactivity.

If the reactor were scrammed at turnaround, the prompt reactivity effect associated with the cooling of the uranium would add to the excess reactivity 0.5 per cent and 1.0 per cent for the present and overbore cases, respectively. Adding this effect onto the startup transient would result in maximum excess reactivities of 4.5 per cent and 4.3 per cent for the overbore and present reactor cases, respectively.

Additional startup transients are shown in Figure 5. The case here is for a minimum outage startup — one in which the reactor startup is made as soon as the shutdown xenon has decayed sufficiently to allow a critical state. The average pile exposure assumed is 400 MW/D/T, which should not be exceeded over most of the reactor operation. Thus, the curves in Figure 5 are more typical of normal operation. The maximum excess reactivities, again assuming a scram at turnaround, are ~ 3.0 per cent and 3.1 per cent, respectively for the overbore and present reactor cases.

The most pronounced differences in the transients are linked to the less-positive, over-all, temperature coefficient and the smaller iodine and xenon concentrations in the overbore reactor. The largest reactivity variation in the overbore case over the entire startup period is ~ 2.0 per cent, which is considerably less than that in the present reactor. The reduced reactivity variation and the smaller maximum excess reactivity in the overbore reactor become important factors in evaluating the over-all safety, particularly since the strengths of the control system are also expected to be reduced. This is further discussed in the section on control systems.

Loss-of-Coolant Reactivity Effect

The present C lattice is not fail-safe upon loss of coolant. That is, the dry lattice is a more reactive configuration than the wet lattice. The magnitude of the reactivity gain on coolant loss is a function of graphite temperature, exposure and fuel geometry. There is also a slight dependence on samarium and xenon concentrations.

The undermoderated, overbore C lattice is fail-safe upon coolant loss under all circumstances. A comparison of the coolant-loss reactivity effects in the two cases is shown in Table VI. The overbore data are based on theoretical calculations; the present reactor data are based on experimental determinations for the cold, green case and theoretical calculation for the hot, ripe case.

Reactivity (Per Cent)

FIGURE 6 STARTUP TRANSIENT

(REACTOR - MINIMUM DETAIL)

(400 MW/1%)

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TABLE VI

Water Loss Reactivity Effects

	<u>C-Present</u>	<u>C-Overbore</u>
Cold, Clean, Green	+3.5%	-0.9%
Hot, $\bar{k} = 400 \text{ MW/D/T}$	+1.6%	-1.9%

Lattice Flooding

In the present C-Reactor, flooding causes a reactivity loss. Addition of water to the undermoderated lattice of the overbore reactor will (under some circumstances) cause an increase in reactivity. In an average cross section of a C-Reactor lattice, there is a void area of approximately 11 cm². (Control rod channels are proportionately divided into each lattice unit). The void area after overboring will not be increased.

In addition, the porosity of the graphite represents an additional void area of about 80 cm². The amount of water which can be absorbed in the graphite blocks themselves is subject to considerable uncertainty since any trapped gases would have to be displaced by the water. Laboratory tests¹¹ have shown that under the most favorable circumstances (either in evacuated graphite or heated graphite) the uptake of water in small graphite samples can approach 75 per cent of the volume available in the graphite. Assuming that this is possible in the reactor if flooding occurred, the available void area to water would thus be 71 cm² per lattice cross section.

In the overbore design assumed in this review, the maximum-conceivable reactivity gain on flooding is calculated to be 0.6 per cent. The peak is reached for a flooded area per lattice of less than 7.0 cm². Thus, it is unnecessary to be concerned about any absorption of water in the graphite itself since additional flooding beyond the 7.0 cm² should result in a reactivity loss. The maximum reactivity gain from flooding occurs in the hot reactor and for green fuel. The calculations assume that the entire reactor is flooded, that water flows from the center of each lattice unit out, and that no water is flashed to steam. Each of these assumptions is conservative.

Maximum Reactivity Added by Increasing Temperature

The maximum reactivity which can be added by increasing temperature can only be estimated since reactivity effects occurring during extreme reactor conditions are very uncertain. However, a fairly reliable comparison of the various effects can be made between two reactor systems, the present and overbore C-Reactor. The items which must be considered are temperature coefficients, loss of coolant, fuel melting, removal of control poisons by melting or vaporization, and chemical and physical changes.

¹¹ R. W. Benoliel, Private Communication.

* This is for the hot reactor; the reactivity gain is 0.2 per cent in the cold, shutdown reactor.

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Temperature Coefficients

The temperature coefficients in the wet reactor have already been discussed. Any unlimited increase in reactor temperature in the reactor must be associated with an interruption in reactor cooling. Thus, the dry pile temperature coefficients are important in the following discussion.

Unlike, the graphite temperature coefficient in the wet pile, the coefficient in the dry reactor is negative at zero exposure. This has been shown in actual startup measurements and is also predicted from theory.^{12,13} One must rely on theory, however, to determine the graphite coefficient at fuel exposures other than zero. The results of calculations of the graphite temperature reactivity transient in the dry pile are shown in Figures 6 and 7. The curves shown are for average pile exposures of zero and 600 MWD/T, which covers the range of normal pile exposures.

The curves point out important changes in the behavior of the graphite temperature coefficient. For zero exposure, the temperature-dependence is reduced in the overbore case again attesting to the smaller effect of temperature on the thermal neutron absorption and multiplication characteristics of the blacker overbore fuel. However, the effect of plutonium-239 is evident at a lower temperature in the dry overbore reactor as the curves for the 600 MWD/T exposure show. At higher temperatures, the exposure-dependence for the overbore case is not so positive, due possibly to earlier competition of higher plutonium isotopes in the neutron balance.

Based on the curves shown in Figures 6 and 7, the maximum reactivity gain due to graphite heating in the dry pile from 20 C to uranium melting at 1133 C is 2.3 and 2.2 per cent for the present C Reactor and the overbore C Reactor, respectively.

The importance of the resonance absorption in U^{238} in the reactor depends on the moderator-to-uranium ratio. The decrease in this ratio which is caused by coolant removal serves to increase the probability of resonance capture and hence the magnitude of the Doppler coefficient. The dry pile Doppler coefficients are given below for the two cases:

$$\frac{1}{k} \frac{dk}{dT} \approx 3.7 \times 10^{-4} (T + 273)^{-1/2} / ^\circ\text{C} \quad \text{C Present}$$

$$\frac{1}{k} \frac{dk}{dT} \approx 7.4 \times 10^{-4} (T + 273)^{-1/2} / ^\circ\text{C} \quad \text{C Overbore}$$

The total Doppler defect in the dry pile to uranium melting are 1.4 and 2.8 per cent for the present C and overbore C Reactors, respectively.

Melting

Following loss of cooling to the reactor, the decay of fission products will release sufficient heat to eventually raise the temperature of the reactor considerably beyond the melting points of the reactor materials. Depending on the rate of heating, when the temperature of the reactor reaches the melting point of aluminum (660 C) either the aluminum cladding on the fuel elements will melt or will alloy with the uranium to form the diffusion alloy UAl₃.

12. HW-37158, K Pile Startup Experiments, Part II, Measured Temperature Coefficients of Reactivity, PT 2, J. H. Brown, June 14, 1955 (Secret).
13. HW-36825, Temperature Coefficients of Heterogeneous Reactors - I, G. W. Stuart, May 25, 1955 (Conf.)

REACTIVITY (PER CENT)

0 1 2 3

1

2

3

FIGURE 6

MODERATOR TEMPERATURE CORRECTION

(TEMPERATURE 10^3 (K/273))

REACTOR CHARACTERISTICS

DATA

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The formation of the diffusion alloy has been previously estimated to result in a loss of local reactivity in the present pile of the order of 2.5 per cent¹⁴. The loss is mainly due to an increase in the capture of resonance neutrons as the uranium-aluminum alloy expands and fills the tube channel volume. The thermal utilization of the fuel is increased and the fast effect is decreased by the change. The new overbore geometry will alter these changes. Based on an analysis similar to that described in Reference 14, the effect will still be a reactivity loss, but of not as large a value. The effect is expected to yield at most a one per cent loss in reactivity. A major difference in the overbore case, assuming the process tubes are of Zircaloy-2, is that there is less aluminum to interact to form the alloy.

For very rapid heating rates, the formation of the alloy (which melts at 1350 C) may not occur and the earlier effect of fuel melting must be considered. In the present reactor, major melting of the aluminum process tubes at 660 C would have undoubtedly preceded any uranium melting and would allow the molten uranium to flow freely. Tests (described in Reference 15) on the effect of various types of fuel redistribution in a cold lattice have shown that a loss in reactivity of 0.5 per cent accompanies fuel slumping in the channel. This loss would be less in the pile at the melting temperature of uranium; here calculations indicate a 0.3 per cent loss. Flowing of uranium into the lattice voids can lead to even higher losses. The measurements referred to in Reference 15 indicated a seven per cent reactivity loss for a simulated complete distribution of uranium in the graphite.

The presence of the relative-high-melting-point, Zircaloy-2 process tube in the overbore reactor will undoubtedly change the sequence of events in a reactor meltdown. The conclusions reached in an earlier hazards review of the use of zirconium tubes in the reactor¹⁶ which were based on then-incomplete meltdown tests have been substantiated by additional work. As reported in Reference 17, additional tests have been made at temperatures up to 1580 C (measured at the interface between the molten uranium and the zirconium tube). In these tests, there was no positive, visual indication of any penetration of the melt through the zirconium. Metallographic examination after the highest temperature tests revealed that the Zircaloy-2 was destroyed in the area of contact with the molten fuel and zirconium was dispersed throughout the uranium mass. Free-flow of uranium following zirconium destruction did not occur, however. It seems safe to conclude that the zirconium tube will contain the uranium melt for some time beyond the time the assembly has reached the melting point of uranium or UAl_3 .

The reactivity effect due to fuel slumping in the overbore reactor will therefore differ somewhat due to the expected confinement of the melt to a relatively smaller channel by the higher-melting point, Zircaloy process tube. Calculations indicate a reactivity gain of on the order of 0.1 per cent for fuel slumping. However, the reactivity loss if the fuel flows into the graphite stack is expected

14. HW-54188, Total Control Problem - Hanford Reactors, J. H. Brown, December 16, 1957 (Secret).
15. HW-50284, Total Control Requirement - Hanford Files, J. H. Brown, May 21, 1957 (Secret).
16. HW-63924, Reactor Hazards Review Zirconium Retubing H Hanford Reactors, R. Nilson, February 15, 1960 (Secret).
17. HW-64573, Observations on a Zircaloy-Uranium-Aluminum System at High Temperatures, A. J. Scott, May 22, 1960. (Unclassified)

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to be higher in the overbore reactor. The reasoning here is as follows: redistribution of the uranium in the graphite stack is in effect a partial homogenization of the reactor. The resulting increase in resonance capture (which produces a reactivity loss) should dominate the changes in the neutron balance caused by the partial homogenization. In the overbore reactor, the greater importance of resonance capture on the neutron balance, should cause a larger reactivity loss from the redistribution of the uranium in the lattice.

If it is assumed that the Zircaloy process tubes in the overbore reactor will actually contain the melt up until the Zircaloy tube itself melts at 1850 C (which is unlikely), then attention must also be given to the possible release of volatile, fission-product poisons or removal of safety control elements from the pile by melting before sufficient uranium redistribution can take place to permanently drive the reactor far subcritical. The evidence of the skull formation, which effectively contained the melt in the melting tests previously discussed, gives some assurance of xenon containment beyond the time the uranium has melted. The boron-steel materials comprising the present vertical safety systems will melt at approximately 1450 C; a temperature above uranium melting, but considerably below zirconium melting. Realistic consideration of loss of poisons in the reactor during a meltdown, however, must also consider their time rates. This is deferred to discussion in later sections.

Summary of Reactivity Effects Associated With Increasing Temperature

The reactivity effects associated with increasing temperature to the melting point of uranium are compared for the two cases in Table VII. The initial condition is the cold, wet, xenon-free reactor with an average fuel exposure of 600 MWD/T.

TABLE VII

Reactivity Effects Associated With Increasing Temperature

	<u>C-Present</u>	<u>C-Overbore</u>
Coolant Loss	+3.5%	-0.9%
Total Doppler Defect to 1133 C	-1.4%	-2.8%
Total Graphite Coefficient to 1133 C	+2.3%	+2.2%
Formation of UAl_3 *	-2.4%	-1.0%
Fuel Slumping*	-0.3%	+0.1%
Fuel Redistribution in Lattice*	-7.0%	-7.0% (at least)

* Last three effects are not additive.

The results in the table show the considerably-greater net reactivity loss which is potentially possible in the overbore reactor due to increasing temperature.

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CONTROL SYSTEMS

There are four, basically-different nuclear control systems in C Reactor. Normal operating control is accomplished with an electrically-driven set of control rods which move in a horizontal direction, perpendicular to the fuel axes of the reactor. This system, by itself, is not adequate to compensate for the total reactivity variation over the reactor cycle and is supplemented by manually-inserted poison strips called splines. These are relatively weak control elements which are inserted into specially-capped process tubes, the splines resting between the ribs of the process tubes. Routine and emergency shutdown control is accomplished by gravity-driven, vertical safety rods. A semi-fluid Ball 3X system provides backup protection to the vertical rods. This system is composed of hoppers located on top of the reactor which open when required and drop small boron-steel balls into the vertical control channels of the reactor.

The design of the C-Reactor control systems was patterned basically after those of the original Hanford reactors with some increase in number of rods and ball channels. However, the same problem of limited geometrical control coverage exists and the reactor loadings are control-limited. That is, there is an upper limit to the amount of enrichment-flattening which can be placed exterior to the control rod patterns and this limit is governed by total control requirements for the maximum positive reactivity accident. (See the section on accident analysis).

The incomplete geometrical coverage of the control systems makes the effective control strengths dependent on the reactor loading and associated flux distribution and in general the pile-values for the control strengths will differ markedly from the local control strengths. The geometry effect on the control strengths should be unchanged due to overboring if the loading as shown in Figure 3 is used. The flux distribution over the reactor for this assumed loading is nearly identical to the present C-Reactor flux distribution.

However, due to the faster neutron spectrum and the reducing of the neutron migration area which the overboring of C Reactor causes, the control system strengths of C Reactor will be reduced by the modification. The ratio of age to diffusion area (τ/L^2) will be increased by a factor of about 1.5. The ratio of epithermal flux per unit logarithmic energy decrement to the total thermal flux will be about twice in the overbored reactor than that which is now characteristic of C Reactor. Since the poisons in the control systems are predominantly thermal absorbers, the faster spectrum results in reduced control strengths. The lowering of the migration area further reduces the reactivity worths of the control systems. The decrease is mainly in the diffusion area; the neutrons diffuse a shorter distance in the overbore reactor and hence the probability of capture in the control elements is reduced.

Two-group, diffusion theory calculations have been made to estimate the control strengths in the overbored reactor. These have been compared to present reactor values with suitable normalization. Part of these calculations are described in References 11-13. The local strengths for various lattice conditions are tabulated in Table VIII. The pile-effective values are given in Table IX. The control strength variation with exposure (due to neutron spectrum changes as U^{235} is destroyed and Pu^{239} and fission products are formed) is small compared to the control strength

11. HW-63080, Local Control Strength Change Due to Overboring, C. E. Bowers and D. G. Montague, December 16, 1959. (Secret)
12. HW-63959, C Reactor Overbore Study, W. L. Bunch, May 15, 1961 (Secret)
13. HW-74046, Effect on C and K Control Capacity of VSR Liners, C. E. Bowers, June 19, 1962 (Secret)

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dependences on temperature, loss of coolant and flooding. The values listed in the tables are thus representative for dirty, ripe conditions as well as clean, green conditions.

It should be noted that the variation in the effective pile values is not as great as the variation in the local control strengths. This is due to the fact that in the reactor, as loaded, the control is 'over-black' in the central regions covered by the control systems, and the main effect of weaker or stronger control elements takes place in the fringe of the reactor, outside of the control region. Thus, whereas there might be a 25 per cent difference between local control strengths between the present and overbored C lattice, the difference in the effective pile strengths may be only half this amount.

TABLE VIII

LOCAL CONTROL STRENGTHS

	<u>Pile Condition</u>	<u>C Present</u>		<u>C Overbore</u>	
		<u>ΔB^2</u>	<u>$\Delta k/k$</u>	<u>ΔB^2</u>	<u>$\Delta k/k$</u>
A. Horizontal Control Rods	Cold-Wet ^a	63 μ b	4.0%	53 μ b	2.8%
	Hot ^b -Wet	68 μ b	4.7%	56 μ b	3.1%
B. Vertical Safety Rods	Cold-Wet	122 μ b	7.7%	102 μ b	5.3%
	Hot-Wet	131 μ b	9.1%	109 μ b	6.0%
	Cold-Dry ^c	109 μ b	7.4%	90 μ b	5.2%
	Hot-Dry	122 μ b	9.4%	97 μ b	6.1%
	Cold-Flooded ^d	-	-	104 μ b	5.0%
	Hot-Flooded	-	-	115 μ b	5.4%
C. <u>Ball 3X System</u>	Cold-Wet	152 μ b	9.6%	127 μ b	6.6%
	Hot-Wet	164 μ b	11.4%	136 μ b	7.5%
	Cold-Dry	136 μ b	9.2%	112 μ b	6.5%
	Hot-Dry	152 μ b	11.7%	121 μ b	7.6%
	Cold-Flooded	-	-	113 μ b	6.2%
	Hot-Flooded	-	-	144 μ b	6.7%

^aWet means normal coolant volume in process tubes

^bHot means 460 C graphite, 50 C coolant (if present)

^cDry means no coolant in process tubes

^dFlooded means water in core (outside process tubes) arranged so as to give maximum reactivity increase.

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TABLE IXPILE-EFFECTIVE CONTROL STRENGTHS

	<u>Pile Condition</u>	<u>C Present</u> <u>$\Delta k/k$</u>	<u>C Overbore</u> <u>$\Delta k/k$</u>
A. <u>Horizontal Control Rods</u>	Cold-Wet	2.5%	1.9%
	Hot-Wet	2.7%	2.1%
B. <u>Vertical Safety Rods</u>	Cold-Wet	4.1%	3.4%
	Hot-Wet	4.5%	3.6%
	Cold-Dry	4.0%	3.4%
	Hot-Dry	4.6%	3.6%
	Cold-Flooded	--	3.3%
	Hot-Flooded	--	3.4%
C. <u>Ball 3X System</u>	Cold-Wet	4.6%	3.8%
	Hot-Wet	5.1%	4.0%
	Cold-Dry	4.5%	3.8%
	Hot-Dry	5.3%	4.0%
	Cold-Flooded	- -	3.7%
	Hot-Flooded	- -	3.8%

Pile Effective Values and Other Information

Additional information on the control and safety systems is given in Tables X and XI. The following will help in understanding the information in the tables, particularly that related to maximum rates at which reactivity can be added.

The horizontal rod system cannot be withdrawn automatically -- manual operation by control handles is required. The maximum credible rate by which reactivity can be added is that resulting from the simultaneous withdrawal of four rods by two operators each using both hands to operate the four available control handles. In the present reactor such an action would add reactivity at a maximum rate of 0.05 per cent per second; the values in the overbore reactor are less due to the reduced strength of the rods.

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TABLE X - CONTROL SYSTEMS - C REACTOR

	<u>C-Present</u>	<u>C-Overbore</u>
<u>Horizontal Control Rod System</u>		
Number of Rods	15	15
Total Reactivity Worth ^a	2.5%	1.9%
Maximum Single Rod Worth	0.3%	0.2%
Rod Dimensions (Poison)	2.84" O.D.	2.84" O.D.
Rod Composition	Boron ^b	Boron
Maximum Rate System Automatically Inserts Reactivity	0	0
Maximum Rate System Permits Manual-Control Insertion of Reactivity	$\pm 0.05\%/sec^c$	$\pm 0.04\%/sec$
Maximum Reactivity Under Control of One Actuator (One rod per control handle)	0.3%	0.2%
Number of Rods to Keep Reactor ^d Down (Cold)	12	> 15
<u>Supplementary Control System (Splines)</u>		
Number of splines (Maximum Required)	90 ^e	170 ^e
Individual Reactivity worth	0.02% max.	$\sim 0.012\%$ max.
Dimensions	0.5" x 0.05" x variable length	0.5" x 0.05" x variable length
Composition	12% Boron sintered into aluminum ribbon	
Maximum Rate of Reactivity Variation	$\pm .0007\%/sec$	$\pm .0004\%/sec$

^aCold, wet reactor. Hot values are ten per cent higher.

^bThere are seven cadmium rods in C-Reactor at present which will be replaced by boron rods by January, 1963.

^cAssuming two operators each using both hands to operate the four available rod control handles.

^dFor green, xenon-and-samarium-free condition. Question is somewhat academic since Standards require reactor be at least 0.15% $\Delta k/k$ subcritical with HCR's in during an outage. Supplementary poison can be charged to meet Standard requirements.

^eAt turnaround for 600 MWD/T average pile exposure.

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TABLE XIa - SAFETY SYSTEMS - C REACTOR

	<u>C-Present</u>	<u>C-Overbore</u>
<u>Vertical Safety Rod System (VSR)</u>		
Number of Rods	45	45
Total Reactivity Worth (Cold/Hot)	4.1/4.5%	3.4/3.6%
Maximum Single Rod Worth ^a	~ 0.3%	~ 0.22%
Poison	Boron Steel	Boron Steel
Rod Dimensions (Poison)	2.98" O.D.	2.98" O.D.
Reactivity Rates (Positive)		
Procedural ^b	0.03%/sec	0.022%/sec
Maximum possible ^c	0.225%/sec	0.18%/sec
Reactivity Rates (Negative)	2.5%/sec	2.0%/sec
Maximum Reactivity Under Control of One Actuator ^d (Hot-Wet)	4.5%	3.6%
Shutdown Margins ^e		
Cold, Clean, Green Pile	+1.6%	+0.5%
Hot Pile	+3.5%	+2.6%
Minimum (Accident) ^f	-1.5% ^g	-0.6%

^aEstimated values in shutdown, hot reactor with maximum flux distortion. Maximum value in normal, operating flux distribution is about 35 per cent smaller.

^bTwo rods at a time - 20 seconds to remove each set of two rods. Distorted flux.

^cAll rods out in 20 seconds. Prohibited by procedures except for hot reactor startups. Rate given for hot reactor.

^dAll rods can be withdrawn with one switch. Withdrawal mechanisms are separate, however.

^eAssuming no other control in reactor.

^fCold, xenon-free dry pile for present reactor; optimum flooding at turnaround (600 MWD/T average exposure) for overbore reactor.

^gA negative shut down margin means the reactor would be supercritical with the specific controls inserted for the specific case listed.

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TABLE XIb - SAFETY SYSTEMS C REACTOR

	<u>C-Present</u>	<u>C-Overbore</u>
<u>Ball 3X System</u>		
Number of Channels	45	45
Total Reactivity Worth (Cold/Hot)	4.6/5.1%	3.8/4.0%
Maximum Single Channel Worth ^a	0.4%	0.3%
Poison	Boron Steel	Boron Steel
Channel Size	4-3/16" Square	4-3/16" Square
Ball Dimensions	3/8" Diameter	3/8" Diameter
Reactivity Rates (Positive)		
Maximum ^b	0.05%/sec	0.04%/sec
Reactivity Rates (Negative) ^c	0.28%/sec	0.23%/sec
Maximum Reactivity Under Control of one Actuator ^d (Hot-Wet)	5.1%	4.0%
Shutdown Margins ^e		
Cold Pile	+2.1%	+0.9%
Hot Pile	+4.1%	+3.0%
Minimum (Accident) ^f	-1.0% ^g	-0.2%

^aEstimated values in shutdown, hot reactor with maximum flux distortion.

^bLimited by speed at which a ball channel can be emptied. Only one channel can so be emptied - ball-handling equipment under reactor can remove only one channel of balls every fifteen minutes.

^cAssuming 16 seconds to fill channels.

^dAll ball hoppers can be opened with one switch.

^eAssuming no other controls in reactor.

^fCold, xenon-free dry pile for present reactor; optimum flooding at turnaround (600 MWD/T average exposure) for overbore reactor.

^gA negative shut down margin means the reactor would be supercritical with the specific controls inserted for the specific case listed.

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The splines used for supplementary control are normally withdrawn by a manually-operated windup reel. The speed of the reel and the small poison worth of the spline restrict the magnitude of any possible reactivity ramp to a very small value. The spline removal process is physically arranged so that only one spline can be withdrawn at a time.

The withdrawal of vertical safety rods presently allows the maximum rate that any control system can potentially add reactivity. Procedures during cold startups limit VSR withdrawal to two rods at a time; this results in the reactivity rates shown in the table. If all VSR's were withdrawn simultaneously (which is now possible) the ramps are considerably larger. Steps are being taken to limit the ramp rate from all rods being withdrawn simultaneously.

The only way the ball 3X system can add reactivity to the pile is if the drain-valves of the channels were to open inadvertently and allow the balls to fall out of the reactor. However, such an event is limited since the ball conveyor system acts as a check on ball removal.

The shutdown margins given for the safety control system in Tables XI (a and b) are calculated assuming that there is no other control in the reactor. This will usually not be the case. For example, in the present C Reactor, sufficient operational control strength in the form of horizontal rods, supplemental poison, or xenon will be maintained in the reactor at all times to compensate for the known excess reactivity. The vertical rods, even though inserted during outages, are not depended on to furnish any of this control capacity. If coolant is lost in this situation, the reactivity increase (about 3.5%) would be controlled by either vertical safety system. There is the possibility, however remote, that a loading or poison-charging error or unplanned control system withdrawal during an outage will result in an undetected reactivity increase which will exceed the strength of the inserted operational controls. If the error is large enough, a simultaneous loss-of-coolant accident would result in exceeding the strength of the vertical safety rods. If the error were still larger, a simultaneous loss-of-coolant accident would result in exceeding the strength of the Ball-3X system. In essence, then, neither vertical safety system is capable of controlling the excess reactivity of the dry reactor unless a large portion of the appropriate operational control system is in place.

The situation in the overbored reactor will be much improved. The lattice will lose reactivity upon loss-of-coolant. In the cold, shutdown, optimally-flooded reactor, either vertical safety system will be capable of maintaining the reactor subcritical without relying on the presence of any of the operational control system. The accident case where neither safety control system would hold down the excess reactivity is the optimum flooding accident at the turnaround peak following a startup from an infinite outage (little or no xenon present). However, it is impossible to attain this condition without operating the reactor with the operational control system inserted so as to maintain a just-critical condition. The excess reactivity during reactor operation must be compensated by control systems to maintain a desired power level. An attempt to withdraw operational controls at this point would result in a power increase obvious to the operator and an automatic scram.

TABLE XII

Nuclear Instrumentation

<u>System</u>	<u>Number of Channels</u>	<u>Range</u>	<u>In Safety Circuit</u>	<u>Sensor</u>	<u>Trip Philosophy</u>	<u>Response to Electrical or Mechanical Failure</u>
Beckman	4	Intermediate	Yes	GE Mark II Ion Chamber	Two-out-of-four will scram; with one bypassed, two-out-of-three. Two bypassed channels causes scram.	Loss of signal operates low trip on controller and behaves as one channel trip. Failure annunciated for both high and low trips. Loss of power causes scram.
Subcritical Monitor and Period Meter	2	Subcritical. Intermediate	No	Fission Counter	Alarm on excessive period.	Chambers are moved by remote drives. Mechanical failure in positioning systems will result in counters not changing position. Erratic behavior will signify certain instrument failures.
Octant	8	Intermediate. Power	No	Ion Chamber	Monitor flux in each octant of reactor.	Erratic behavior signifies instrument failure.
Proportional Counter	2	Subcritical. Intermediate	No	BF ₃ Proportional Counter	Reactor period manually calculated from count rate.	Erratic behavior signifies instrument failure.

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TABLE XIII

Temperature Monitoring Systems

<u>System</u>	<u>Number of Channels</u>	<u>Range</u>	<u>Sensor</u>	<u>Response to Failures</u>	<u>Remarks</u>
Power Rate of Rise	10	Intermediate Power	Resistance Thermal Detectors on Ten Tubes	Erratic behavior	Not in service at present.
Outlet Temperature		All	Thermocouple	Erratic or inoperative behavior	Used for periodic scanning and recording of outlet tube temperatures. Hot tubes identified.
Twenty-Tube Average Outlet Temperature Monitor	20	All	Thermocouple	Erratic behavior	Used to obtain rapid indication of power changes. Signals from 20 TC's are averaged.
Twelve-Point Recorder	12	All	Thermocouple	TC failure gives downscale reading	Used to indicate small changes in outlet temperature of 12 selected tubes.
Bulk Exit Water Temperature System	2	All	Resistance Thermal Detector	Erratic behavior	Measures bulk outlet water temperature to monitor against boiling.
Inlet Water Temperature Monitor	9	All	Resistance Thermal Detector	Erratic behavior	Used to determine ΔT across reactor to calculate power.

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TABLE XIV

Pressure and Flow Monitoring Systems

<u>System Variable</u>	<u>Number of Channels</u>	<u>In Safety Circuit</u>	<u>Sensor</u>	<u>Philosophy of Operation</u>	<u>Remarks</u>
Process Tube Coolant Flow	1980	Yes	Bourdon Tube	1/1980 on low or high flow	Monitors flow of coolant to each process tube. Highly fail-safe.
Inlet Riser Pressure	4	No	Bourdon Gage		Monitors riser pressures.
Inlet Riser Pressure	6	Yes	Mercoild Switch	IP and IIP Time Delay	Secures reactor on loss of inlet pressure. If Δt between IP and IIP trip short, Ball 3X trip also.
Inlet Header Pressure	6	No	Foxboro Gage		Monitors header pressures
Cross Header ΔP	46	No	Barton ΔP Gage		Monitors ΔP across unit -- between front and rear cross headers at shutdown only.
ECR Coolant Flow	15	No	Bourdon Tube		Monitors coolant to horizontal rods
Thermal Shield Coolant Flow	1	No	Mercoild Pressure Gage		Monitors coolant to shields; also used in power calculator
High Tank Pressure	2	No	Foxboro Bourdon Tube		Monitors high tank (2) pressures
Raw Water Pressure	1	No	Mechanical Gage		Monitors raw water supply pressure

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Thus, in spite of the negative shutdown margins of the vertical safety systems, if they were the only control poison on the flooded reactor at turnaround, the situation in which they become the only control elements present is not possible with normal operating procedures and probabilities for errors.

INSTRUMENTATION

No changes in reactor instrumentation, except for recalibrations and revisions in certain trip settings, are planned in the modification. The basic reactor instrumentation systems are summarized in Tables XII, XIII, XIV. There is no incore instrumentation.

Detection and location of failed fuel elements is accomplished by a gamma-monitoring system. Coolant is sampled from each rear crossheader and rear riser, a total of 94 sample points. There are eight sensing heads, each scanning 12 samples.

Other instrumentation not included in the tables are

1. High tank water level.
2. Reactor thermocouples to measure graphite temperatures (259 couples plus 22 thermocouple stringers).
3. Horizontal-rod water temperature.
4. Thermal-shield temperature (125 thermocouples)
5. Seismoscopes. Three pendulums, one sensitive and two less sensitive. Trip of sensitive circuit causes alarm; trip of both less-sensitive circuits causes scram.
6. Power calculator.
7. Gas activity monitor.
8. Gas flow, pressure and temperature monitors.
9. Gas analyzers (helium, oxygen)
10. Dew point monitor for pile atmosphere.

CONTAINMENT

The Hanford reactors are not contained. The overbore project does not include containment. Recently, however, the Hanford reactor buildings have undergone a confinement modification which provides some confinement of fission products. The salient features of the modification are

1. Filters in the exhaust air channel from the reactor buildings.
2. Fog spray system for rear face working area.

EMERGENCY SYSTEMS

Criteria* for Hanford reactor cooling systems state that there shall be three separate and independent cooling systems (some common piping and pumps permissible), each of which shall be capable of providing adequate reactor shutdown cooling. The failure of one system should not prevent the others from operating.

*See Appendix A.

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For C Reactor, the primary coolant system consists of ten primary pumps operating on BPA power. The secondary coolant system consists of the primary pump flywheels and the steam-turbine driven backup pumps. A third, or last-ditch, coolant supply system is required to provide adequate reactor cooling in the remote event of the combined loss of primary and secondary water supply to the reactor. This system now consists of two high tanks and the export system that interconnects pumping stations at each of the four old reactor areas and the inlet manifolds of the six older reactors. In the case of a complete electric and steam pump power failure, the primary process pump flow decays down, and high tank flow takes over about three minutes following the loss. These high tanks serve the purpose of providing a temporary reactor coolant flow of intermediate rate to fill the gap between the high flow from the primary pumps and the low flow from the export system. The export system provides reactor flow within twelve minutes following failure of reactor pumping power. It adds to the high tank flow and extends the time required to drain the high tanks. When the high tanks are empty, the export system is the sole source of cooling. (As noted in the introduction, the project also calls for improving the reliability of the last-ditch system by replacing the export system by automatic, fail-safe, pumping systems which will be independent of all other coolant systems).

A comparison of the adequacy of C Reactor backup cooling systems under present and overbore conditions is shown in Figure 8. The top curves show the flow available and the bottom curves the flow required to maintain a 90 C bulk outlet temperature as a function of time after power loss. The points of interest are (1) the point where the high tank flow takes over, and (2) the point where the export system is the sole supply of coolant. The high tank flow in the overbore case is 1200 gpm greater because of the lower hydraulic resistance of the overbored reactor. However, this increased flow rate causes the high tanks to drain about six minutes earlier in the overbore case than in the present case. A necessary export flow of 7000 gpm shown in Figure 8 is not currently available. However, the system is currently being modified to provide this flow rate.*

The coolant flow required to maintain a 90 C bulk outlet temperature is shown in the bottom curves. These curves were calculated on the basis of a theoretical, initial, graphite heat-generation contribution of 4.5 per cent for overbore versus six per cent for the present reactor. It was also assumed that the graphite relaxation period would increase by 30 minutes due to the decreased thermal conductivity of the zirconium tubes as compared to aluminum, and a possible change to a gas composition with greater resistance to heat flow.

As shown in Figure 8, the calculations indicate that the present backup system with modified export is adequate for the overbored reactor.

PLANNED DEVELOPMENT PROGRAMS

Although the overbore proposal is a modification of an existing Hanford reactor, certain development problems do exist. The critical areas are fuel performance, control adequacy and hydraulic considerations. The following development program is planned and some work is already underway.

*Plant Project CGI-905, "Interim Modifications for Improved Coolant Backup, B, C, D, DR, F and H Reactors".

DECLASSIFIED

-39-

HW-72374

Time After Power Loss - Seconds

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Fuel Performance

Sixty-two channels in C Reactor have been overbored and are serving as test columns to irradiate prototypical overbore size fuel elements under typical overbore power generation and hydraulic conditions. Overbore fuel elements have already been under irradiation for as long as six months. Information to be obtained will be resistance to fuel rupture, fuel-failure mechanisms, dimensional instability, corrosion rates and potential for cleavage failures.

Hydraulics

Steady-state and transient flow studies have been completed to determine the hydraulic and heat transfer characteristics of an overbore column. These studies confirmed the computed channel hydraulic characteristics and fuel design, and provided data to be used as a basis for establishing flow monitoring procedures.

Physics

The material buckling of the overbored lattice will be determined for natural uranium and uranium enriched to 9.947 w/o U^{235} to assist in determining the reactor loading. Control rod and Ball 3X strengths will be measured in the overbored lattice and compared to measurements in the present C lattice. The reactivity effect of water loss and core flooding will be determined. The conversion ratio is being determined by irradiating prototypical elements in the C-Reactor overbore test block.

Materials

The results of relatively short-term irradiation tests indicate that no major deleterious effects should occur during the intended twenty-year life of the Zircaloy process tubes at C Reactor conditions. The longest Zircaloy exposure to date has been seven years in D Reactor with the bulk of the data obtained during the last two years of exposure.* From radiation damage data obtained to date there is no reason to believe that twenty-year life cannot be obtained. It will be prudent, however, to monitor the performance of the Zircaloy tubes throughout their period of use. To accomplish this, tubes will be periodically removed from the reactor to evaluate long-term mechanical, physical, and chemical changes.

Studies will also be directed towards determining the graphite heat generation rates and temperature distribution in the graphite blocks. The overbore program will require a re-evaluation of the effects of distortion, stored energy and graphite burn-out on safety and operational performance.

HAZARDS ANALYSES

A detailed hazards analysis is not presented in this preliminary report. Rather, the significant features differentiating the overbore and present C Reactor designs with respect to nuclear safety are discussed.

*There has been one Zircaloy tube in D Reactor for seven years. There have been about 70 Zircaloy tubes (above the 62 overbore channels) in C Reactor, the majority of them placed in service in 1960.

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Reactor Control and Safety Philosophy

The philosophy of reactor control and safety in the Hanford reactors had its origin in the days of the Manhattan Project. The original philosophy has satisfactorily stood the test of time and has carried on through the years with few changes. Certain changes were made in the philosophy in the Design of N Reactor¹⁴; these changes were in the direction of improved safety. For example, an important criterion for the N-Reactor lattice design was that it be fail-safe upon loss of coolant. To a degree, significant improvements in reactor safety philosophy should be applied to any new Hanford reactor design or significant modification, such as the overbore modification. This is not to say, that the present Hanford reactors must conform in all respects to the N-Reactor safety philosophy, but rather where possible such conformance is desirable and certainly should be considered as an important measure of any major reactor modification.

Operational Control Criteria

The operational control system (horizontal control rods plus any supplemental controls) must meet the following requirements:

1. Adequate strength to maintain the reactor subcritical under all normal and shutdown conditions.
2. Adjustment at rates fast enough to override all normal, operational reactivity transients and to minimize delays at startup. However, the systems must be limited in their withdrawal rates to that rate which, in coincidence with simultaneous failure of the Primary Safety System (vertical safety rods), would not result in any fuel melting.
3. Precision positioning and movement to permit establishing and maintaining the reactor power level within desired limits.
4. Adequacy of geometrical coverage and flexibility to permit attainment of adequate power-flattening efficiencies and satisfactory control over spatial power cycling.

As far as Criterion 1 is concerned, the horizontal rods, as supplemented by poison splines, have adequate control strength in either the present or overbored C Reactor to override all operating reactivity transients. It is true that the control rods are weaker in the overbored reactor thus necessitating increased usage of poison splines, but such usage of supplemental poisons has now been well established at Hanford and procedures are simple, and offer no major safety problems.

Criterion 2, insofar as the withdrawal rates are concerned, is better satisfied in the overbored reactor due to the weaker rods and the almost-double prompt negative temperature coefficient in the modified reactor. Transient calculations under the assumption of maximum control rod withdrawal rates without an automatic scram have shown that boiling occurs at about six seconds and 11 seconds after

14. HW-72719, "Reactor Control and Safety Systems - NPR," W. A. Burns, et al, February 19, 1962 (Confidential).

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HW-72374

the initial rod movements for the present and overbore reactors, respectively. In the present reactor case, boiling would accelerate the reactivity ramp due to the positive void coefficient, whereas in the overbore case instigation of boiling would tend to quench the reactivity transient and delay fuel melting long enough so that a manual scram could be effected, either by safety rods or by Ball 3X.

Criterion 3 and 4 are no problem in either reactor design. Xenon cycling should be much less of an operational problem in the overbore reactor due to a lower iodine-to-xenon ratio and the smaller over-all temperature coefficient. The reactor should be improved against cycling instabilities.

Safety Control Criteria

Hanford safety philosophy calls for two safety control systems. One of these systems, depending on the nature of the accident, is to provide the primary protection; the other functions as a backup. Each safety system must meet the following requirements:

1. Adequate strength (if inserted) to maintain the reactor subcritical for all credible* accidents at least until the reactor is damaged beyond repair as a result of the causative accident. Beyond this point the safety control system is required to maintain control only to the extent that the consequences of the incident are no more than those which would result from the causative accident alone.
2. Speed-of-control insertion adequate to (a) limit a power excursion from credible accidents which would themselves result in damage (loss of coolant, for example) to one of a magnitude such that the ultimate consequences of the excursion are no worse than those which would have resulted from the causative accident alone, or (b) limit a power excursion from other credible accidents (such as fast rod withdrawal) to one of a magnitude such that no damage beyond mild fuel stressing will result.
3. Physical limitation of withdrawal rate so that (a) a properly-effective scram would limit any power rise (occurring because of the withdrawal) to one of a magnitude such that no damage beyond mild fuel stressing would result or (b) a failure to automatically scram the reactor in coincidence with the maximum withdrawal rate would allow time between a first indication of the excursion and fuel melting to effect a manual scram.

The first criterion is a re-statement of the Hanford Total Control Criterion. In the present C Reactor, this criterion is met by supplementing the control strength of the Ball 3X system with poison splines - the accident dictating total control requirements is loss-of-coolant. It is to be pointed out that the total control criterion cannot be met by the vertical rod system under certain water-loss circumstances; hence only one of the safety systems (the Ball 3X) is able to satisfy the Total Control Criterion under certain circumstances.

*Credible accidents exclude acts-of-war, sabotage, and earthquakes so severe that entry of both safety systems into the reactor is prohibited.

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HW-72374

Since, in the overbored reactor, loss of coolant results in a loss of reactivity, meeting total control requirements with either safety system in the dry reactor is no problem. The shutdown reactivity margins are 2.0% and 2.4% for the Vertical Safety Rods and Ball 3X system, respectively, with no supplemental poison required. The optimum flooding accident sets the limit for total control in the overbore reactor. Again, as previously discussed, both safety systems are strong enough, in conjunction with poison already in place in the reactor, to satisfy the criterion. The presence of zirconium process tubes in the overbored reactor will undoubtedly alter the sequence of events in the reactor if coolant is lost and a meltdown occurs. Further study is required to assess the final reactivity state in the reactor if control poisons melt and flow out of the reactor before the reactor is permanently driven subcritical by uranium flowing into the lattice. Previous analyses on other reactors (for example the E-N load and NPR) have shown that even if poison elements do melt, they cannot flow out of the reactor fast enough to produce an excursion which would add significantly to the fission product inventory. The meltdown may be hastened, but meltdown is inevitable anyway. Furthermore, major fission product release would not occur prior to process tube blockage since at or about the melting point of aluminum the diffusion alloy formed between uranium and aluminum would very likely block the tubes.

The second criterion is a re-statement of the Hanford Speed-of-Control Criterion. The speed of the vertical safety rods must now be fast enough to prevent any uranium melting during the power excursion associated with rapid loss of coolant. The excursion which will dictate safety rod response in the overbore reactor is that associated with the maximum withdrawal of the safety rods; this is a ramp at most a tenth as large as that associated with coolant loss in the present reactor. With present VSR withdrawal rates maximally possible, some boiling would take place in a startup accident in which the entire VSR system were withdrawn at their maximum rate. This is not the case in the overbore reactor even with existing instrumentation since, in the first place, the maximum ramp is smaller (weaker rods) and secondly the damping due to Doppler heating is larger. However, if the reactor fails to scram automatically for such a maximum startup accident, neither the present nor the overbore C Reactor could escape without significant boiling occurring from the excursion. The excursion would very likely proceed to fuel melting in the present reactor due to the 'boot-strapping' reactivity effect from coolant loss. In the overbore reactor the negative void coefficient would tend to terminate the excursion.

The third criterion is similar to the one for the horizontal control rods and the previous remarks apply.

Struck Rod Criterion

Although there are relatively few critical masses in the Hanford reactors by virtue of their rather low excess reactivities, there are certain accident situations in which there is a large-enough increase in reactivity so that localized supercritical regions in an otherwise subcritical reactor must be considered. Reactivity balance calculations prior and following a reactivity increase are not

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HW-72374

sensitive enough to determine the now many safety rods (or Ball 3X channels) can be inoperable. Since, the safety control systems physically enter the reactor at the top and take a finite time to be completely inserted, and further since safety circuit sensing apparatus (ion chambers, etc.) are also located, at present, in a poor geometry, it is necessary to study the effects of certain control elements being out of service by solving the space-time dependent reactor kinetics equations. Such calculations effectively set upper limits to the number of stuck or inoperable control elements. In the present C Reactor these calculations have shown that an operable rod must have no more than two adjacent-rods out of service to meet the speed-of-control criterion upon loss of coolant. Process Standards state that there can be at most two inoperable rods and they shall not be adjacent. In the overbore reactor, the accident control demands are less and the stuck rod criterion should not be as restrictive. No transient calculations have been made, however, to determine the limit.

Maximum Credible Accident

The maximum credible accident which is proposed as the basis of the design is identical to that now assumed, namely the simultaneous failure of both the primary and secondary coolant systems. The last-ditch coolant system is therefore relied on to supply cooling to the reactor. As stated earlier, the modification includes a new last-ditch system will be more reliable than the present export system.

Damage Appraisal For Worst Conceivable Accident

The worst conceivable accident is assumed to be an earthquake so severe that the reactor building is shaken down, all coolant lines are broken and the reactor is destroyed by burning. Essentially the full inventory of fission products is released.

No quantitative analyses have been made of the sequence of events in such an accident. The earthquake assumed would be severe enough to prevent reactor shutdown by safety rod insertion and also probably by Ball-3X insertion. In the case of the present reactor, the possibility exists for the reactor to be supercritical if coolant is lost and the reactor fails to scram. However, what additional inventory in fission products a post-accident supercritical state would produce has not been analyzed in any quantitative terms. In the overbore lattice, the loss of coolant is expected to render the reactor subcritical even if the scram system fails thus leaving little likelihood of additional fission products being produced following the accident.

For present purposes, in lieu of any quantitative analyses, it has been considered sufficient to merely compare the fission product inventories in the present and overbored C Reactor under the likely operating conditions and use this as a relative measure of the potential damage from the worst conceivable accident. The data are presented in Tables XV and XVI. The figures for a K Reactor are included for comparison.

The data in the Table XV are given in megacuries and are for 24 hours after reactor shutdown. The exposures and power levels assumed are listed in the footnote in the table. It is noted that the inventory of biologically-significant radioisotopes is increased by about twenty per cent in the C Reactor if the reactor is overbored. However, the inventory is still well below that of a K Reactor.

The longer exposure effects do not show up in the results in Table XV since many of the short-lived fission products (which are primarily a function of power level) are still strong emitters only 24 hours after reactor shutdown. The inventory of the biologically-significant radioisotopes one year after shutdown is given in Table XVI. Here the longer exposure of the overbore fuel is significant in its effect; however, the inventory is still about equal to that from a K Reactor. The exposure and power level assumptions are the same as for Table XIII. The inventories are again given in megacuries.

TABLE XV

Inventory of Biologically-
Significant Fission Products^a
(24-Hour Cooling)

	<u>Inventory (Megacuries)</u>		
	<u>C-Present</u>	<u>C-Overbore</u>	<u>K</u>
<u>Noble Gases</u>			
(Xe ^{133,5})	148	160	272
<u>Halogens</u>			
(I ^{131,2,3,5})	168	188	309
<u>Volatile Solids</u>			
(Ru ^{103,5,6} , Te ^{127,9,132} , Cs ¹³⁷)	122	144	265
<u>Non-Volatile Solids</u>			
(Sr ^{89,90} , Zr ⁹⁵ , Ba ¹⁴⁰ , Ce ¹⁴⁴)	251	334	462

^a 2175 MW for 81 days for C present, 2310 MW for 163 days for C-overbore, 4400 MW for 81 days for K.

From the standpoint of short-term radiation effects, the inventory of biologically-significant radioisotopes in C Reactor will be increased by about twenty per cent, but still will be well below that of a K Reactor. From a long-term radiation standpoint (damage to crops and land), the burden of fission products will be increased by about 90 per cent in C Reactor. This is comparable to that from a K Reactor now.

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TABLE XVI
Inventory of Long-Lived, Biologically-Significant
Radioisotopes One Year After Shutdown

<u>Isotope</u>	<u>C-Present (Mc)</u>	<u>C-Overbore (Mc)</u>	<u>K (Mc)</u>
Sr ⁸⁹	4.2×10^{-1}	6×10^{-1}	7.7×10^{-1}
Sr ⁹⁰	5.64×10^{-1}	1.2	1.0
Zr ⁹⁵	1.37	2.1	2.5
Ba ¹⁴⁰	3×10^{-7}	3.6×10^{-7}	5.5×10^{-7}
Ce ¹⁴⁴	7.7	15	14
Ru ¹⁰³	7.5×10^{-2}	1×10^{-1}	1.4×10^{-1}
Ru ¹⁰⁶	5×10^{-1}	9.8×10^{-1}	9.2×10^{-1}
I ¹³¹	1.7×10^{-13}	1.8×10^{-3}	3.1×10^{-13}
Te ¹²⁷	3×10^{-2}	5×10^{-2}	5.5×10^{-2}
Te ¹²⁹	5×10^{-3}	6.4×10^{-3}	9.2×10^{-3}
<u>Cs¹³⁷</u>	<u>6.15×10^{-1}</u>	<u>1.3</u>	<u>1.1</u>
Total	11.3	21.3	20.5

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APPENDIX A

HANFORD COOLANT SUPPLY CRITERIA

The requirements for the reactor coolant system are defined by the following criteria:

The normal reactor cooling system shall consist of three independent, reliable sources of cooling as follows:

1. Primary Cooling System

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

2. Secondary Cooling System

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

3. Last-Ditch System

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

An independent system is defined as one with power sources and pumping systems such that failure or cause of failure in one cannot induce failure of the alternate systems.

Adequate cooling is defined, as a minimum, as that required to maintain reactor components, including fuel elements, at temperatures below the melting point. In practice, the objective of requiring levels of cooling capable of protecting against all reactor of fuel element damage will be prudent to adopt.

Failure of any single component in a system shall not be capable of rendering that system inadequate. In practice certain piping and valves may be qualified as exceptions.

All engineering, construction, operation and maintenance shall meet or exceed requirements specified by applicable codes and standards.

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