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CURRENT STATUS OF H REACTOR TOTAL CONTROL

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CURRENT STATUS OF H REACTOR TOTAL CONTROL

Introduction

This document presents the total control aspects of the current blanket and central zone E-N loading at H Reactor based on the calculational parameters in reference /1/.

The original charge for the E-N core load at H Reactor, (19.37" E to 1" N) was found to be more reactive than desirable for maximum production efficiency.^{/2/} Therefore a less reactive E to N ratio (18.14" E to 1" N) has now replaced the original loading.

/1/ HW-69566 Physics of E-N Load Compared to Natural Uranium Load at H Reactor,
D.I. Monnie, May 11, 1961

/2/ HW-70740 Recommended E-N Charge Modification (Second Cycle PT IP-350-C E-N
Demonstration Load), R. D. Carter, August 15, 1961

Summary

The E-N fuel is calculated to be slightly more reactive in the dry case than natural uranium with enrichment, but the safety system at H Reactor is adequate for normal shutdowns. In the initial core loading neither the VSR nor Ball 3X control systems would completely suffice for total control during an infinite outage; therefore, the reactivity of the fuel in the flat zone had to be reduced during outages by supplementary control poison. The graphs on Figure 3 show the amount of poison required to reduce effective material buckling in the central zone sufficiently to meet total control requirements.

The modified E-N fuel loading is still calculated to be slightly more reactive in the dry case than natural uranium with enrichment but less than the original loading ratio. Supplementary control poison will be required during periods that the VSR's are in service in place of the Ball 3X system. However, no supplementary control poison will be necessary when the Ball 3X system is not locked out.

Discussion

I. Initial Core Loading

A. Details of the E-N Loading

Charging of the first of two cycles of a tritium-plutonium producing fuel loading (E-N load) was completed at H Reactor on May 28, 1961. The striped columns of "black" lithium-aluminum (Li-Al or N) and enriched (.947% U-235 or E) uranium slugs in the flat zone are surrounded by a blanket Li-Al load supported by E columns.

The theoretical reactivity evaluation of this loading was calculated and documented by D. I. Monnie.^{/1/} The total control aspects of the fringe poison or blanket loading were covered by the author in a previous document.^{/3/}

B. Reactivity Evaluation

The cold dry reactor has the highest reactivity potential because as the dry pile heats, the combination of reactivity effects will reduce reactivity as illustrated in Figure 1. The reactivity at meltdown (aluminum melting temperature) never exceeds the cold-dry reactivity so that the poison charged to keep the cold-dry pile subcritical will also suffice to control any reactivity transients that occur between minimum and meltdown temperature for all contemplated fuel exposures.

C. Reactor Parameters

The E-N fuel loading is illustrated in Figure 2.

A charge length ratio of 19.37 inches E to 1 inch N was calculated to match the lowest reactivity state of the natural uranium and enriched

^{/1/} HW-69566 ibid

^{/3/} HW-68250 Total Control Aspects of the Fringe Poison Loading, A.D. Vaughn
January 26, 1961

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fuel loading. The theoretical physics parameters were calculated for 20 inches E to 1 inch N. Use of these parameters for total control calculations will be conservative because the reactivity becomes greater with an increase of enriched fuel per mint charge.

The relative parameters were taken from reference /1/ and are included in Table 1.

Table I

Calculated Reactivity Parameters

	<u>Cold Dry</u>		<u>Hot Dry</u>	
	<u>I&E Natural Uranium</u>	<u>E-N, Length Ratio 20"/1"</u>	<u>I&E Natural Uranium</u>	<u>E-N Length Ratio 20"/1"</u>
k_{∞}	1.0867	1.1139	1.0699	1.0779

D. Assumptions

The parameters in Table 1 are merely relative values. In order to find true parameters which may be used in total control calculations, normalization factors are necessary. For example, the material buckling for any fuel can be estimated by

$$B^2 = \frac{k_{\infty} - 1}{M^2} \quad \text{Where } M^2 = T + L^2 \text{ as shown in Table II.}$$

Table II

Values of Diffusion Length and Fermi Age
In 8 3/8" Lattice

	<u>For E-N Fuel</u>	<u>For Natural Uranium Fuel</u>
Fermi Age (T)	334 cm ²	334 cm ²
Diffusion Length (L ²)	301 cm ²	310 cm ²

The reactivity parameters listed in reference /1/ for the various fuels are higher than experimental values. The calculated material buckling for natural uranium would be 131 μ b in the dry pile and 90 μ b in the wet pile. The experimental, measured values are 101 μ b and 60 μ b respectively.

The calculated cold clean, wet buckling for the present E-N fuel is listed as 137 μ b. This value was reduced by the equivalent reactivity effect of the supplementary control poison charged before startup, and used (in a one group diffusion theory calculation) to predict critical. The pile actually went critical measuring approximately 25 mk of excess reactivity; the calculation gives a value of 50 mk or 80 μ b of excess reactivity.

Because all the calculated values are unreasonably higher than the measured values, the E-N fuel (i.e., 175 μb) was conservatively normalized to 150 μb . Total control requirements calculated using the unrealistic non-normalized value would be prohibitive.

Because of the reduction in effective pile size due to the blanket loading of fringe poison, additional E columns are necessary to restore adequate operating reactivity, and compensating enrichment has been charged in the adjacent (second to fourth) lattice units. Manipulation of a single mint lattice charge and a nebulous effective pile size would render the answers somewhat unreliable. By experimental evidence (i.e., test pile analysis of fringe poison loadings and previous operating experience) it is concluded that two and one-half lattice units of enriched fuel charged adjacent to the blanket Li-Al will approximately match the wet reactivity effect of the displaced natural uranium and enriched fuel fringe loading.

Since reference /3/ concludes the dry reactivity effect in the fringe (i.e., blanket Li-Al and compensating enrichment) to be no greater than if the same zone were loaded with natural uranium, provided that maximum fringe tube powers do not exceed those in the central zone, the following assumptions can be made:

1. The effective pile size will be the same as if the fringe contained natural uranium.
2. The first three and one-half lattice units from the reactor edge will be treated as having a buckling equivalent to dry natural uranium.
3. Poison charged in the center of the pile will be counted over and above the effect of the top and bottom enrichment, due to the high enrichment concentration in the central zone.

E. Total Control Strength of Safety System

The equation derived by J. A. Wheeler/⁴/ for calculating local control strengths gives the following local control strengths for the VSR's and Ball 3X at H Reactor:

Table III

Calculated Local Control Strengths

	<u>Natural Uranium Loading with Enrichment</u>	<u>E-N Fuel Loading 20" to 1" Length Ratio</u>
VSR's	170 μb	160 μb
Ball 3X	207 μb	181 μb

In actual practice, evaluations based on startup test results have been used. Bucklings actually assumed in the natural loading and their normalized values in the E-N load are shown in Table IV.

/3/ HW-68250, ibid

/4/ N-2292 Principles of Nuclear Power, Chapter 22: Control, p. 77.

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Table IV

Local Control Strengths for Safety Calculations

	<u>Natural Uranium Loading with Enrichment</u>	<u>E-N Fuel Loading 20" to 1" Length Ratio</u>
VSR's	120 μ b	115 μ b
Ball 3X	150 μ b	130 μ b

F. Calculations

1. Preliminary Startup Conditions

Evaluation of safety system capacities by the square pile approximation/^{5/} gives a measure of the system strength in terms of number of enrichment columns allowed. The first step in this procedure is to tabulate the effective bucklings in a three region reactor.

- Region 1. A central zone containing safety controls.
- Region 2. An enriched fuel zone containing controls.
- Region 3. A fringe zone without controls.

The effective bucklings for each zone are as follows:

Cold dry material buckling 20" E/1" N length Ratio = 150 μ b

Buckling increase due to top and bottom enrichment/^{6/} = + 22 μ b

Average material buckling reduction in the flat zone due to 24 spike mint columns and 70 splines (equivalent to 46.2 spike poison columns) = -65 μ b

Reduction in buckling allowed for vertical leakage = -8.5 μ b

Average material buckling in Region 1 = +98.5 μ b

Average material buckling in Region 3 = 101.0 μ b
(Reference HW-68250)

Before applying the square pile theory, the center slab section (called B section) in the oblong reactor must be proven to be a neutron sink. Then the pile size can be reduced to a square pile geometry (cylindrical coordinates), thus facilitating total control calculations.

/5/ HW-62884 Hanford Safety System Capacities by Square Pile Approximations, G.R. Parkos, December 4, 1959

/6/ This is actually the buckling which must be subtracted from the central zone in order to compensate for the top and bottom enrichment effect. This number is more correct since we are calculating the buckling in the central zone.

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The "B" section is divided into two regions, one region outside the safety system, and the other in the safety system pattern with bucklings as follows:

For central B inner section (buckling inside the safety system region with VSR's inserted
= 98.5 - 115 for VSR's = -16.5 μ b

Calculated buckling required in central B outer section (outside safety system region) for the center slab to be just critical = 210 μ b

Therefore, the central "B" section satisfies total control requirements, provided VSR's are inserted, with 70 splines and 24 poison columns charged.

For the three square pile regions the parameters are as follows:

$$B_1 = -16.5 \mu b \quad (98.5-115)$$

$$B_3 = 101 \mu b \quad (\text{dry material buckling of natural uranium})$$

$$B_2 \text{ allowed} = 125 \mu b \quad (\text{calculated value necessary for the dry, square pile to be critical})$$

$$125 \text{ plus VSR strength} = 240 \mu b$$

Fraction of enriched tubes allowed in Region 2 (3.5 lattice units wide and 37 lattice units high) = f

$$240 = f (320) + (1-f) 101 \text{ where } 320 = \text{material buckling of enriched fuel} \quad f = .635$$

$$\text{Number of E columns allowed} \quad 82 = f \times 3.5 \times 37$$

This is the number of E columns allowed on each side of the pile in support of the flat zone. Since the actual number of E columns in this zone is 70 near and 65 far side, the VSR safety system was sufficient to satisfy total control requirements during the initial pile loading.

2. Exposure Variations and Total Control

Calculations by D. I. Monnie indicate a reactivity loss due to enriched fuel burnout as exposure increases. This reactivity decrease has not been evident as the actual fuel loading gains exposure. Therefore, to be conservative, the only reactivity decrease assumed in the total control calculation was the amount caused by samarium buildup. Subtracting the reactivity loss due to samarium, from the cold, green material buckling gives a value greater than any reactivity calculated at aluminum melting temperatures, (see

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Figure 1). Since the samarium transient will reach equilibrium at approximately 150 MWD per ton, this calculation will be applicable for all exposures beyond this.

Cold dry material buckling 20"E/1"N length ratio	= +150 μ b
Buckling decrease due to samarium buildup	= - 10 μ b
Buckling increase due to top and bottom enrichment	= + 22 μ b
Buckling decrease due to (17) equivalent spike mint columns (i.e., 10 spike mint plus 25 splines)	= - 26 μ b
Buckling decrease because of vertical leakage	= <u>- 8.5 μb</u>
Total	127.5 μ b

If VSR's are inserted $\rho_{Bc} = -115 \mu b$

For central B section, buckling inside the safety system region = + 12.5 μ b

Buckling allowed outside the safety system region = 105 μ b

Therefore center "B" section meets total control requirements.

For square pile

$$B_3 = 101.0 \mu b$$

$$B_1 = +12.5 \mu b$$

$$B_2 \text{ allowed} = +55 \mu b + 115 \text{ for VSR's} = 170 \mu b$$

Fraction of enriched fueled tubes in region 2 (3.5 lattice units wide and 37 lattice units high) = f

$$170 = f (320) + (1-f) 101$$

$$f = .310$$

Number of E columns allowed per side = 40 = f x 3.5 x 37
Additional poison was changed until Ball 3X was returned to service.

With the Ball 3X safety system in service the square pile parameters are:

$$B_3 = 101.0$$

$$B_1 = -2.5 = (127.5 - 130)$$

$$B_2 \text{ allowed} = 100 \text{ plus } 130 \mu b \text{ for Ball 3X local control strength}$$

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$$230 = f \cdot 320 + (1-f) \cdot 101$$

$$f = .589$$

$$\text{Number of E columns allowed} = 76$$

II. Second Core Loading

The reactivity parameters relative to natural uranium for E-N ratios of 17.75" E to 1" N and 20.0" E to 1" N were calculated by D. I. Monnie.^{/7/} As mentioned previously, a normalized value of the 20" E to 1" N parameter was used for the initial loading's total control calculations. The reactivity of the modified loading is between the values listed in HW-69566.

Data^{/8/} from a recent experiment at the Hanford Test Pile, which measured the reactivities of various E to N charge length ratios, is sufficient evidence that a linear extrapolation is warranted between the two values calculated by D.I. Monnie. This extrapolation gives a cold, clean, dry material buckling of 140 μ b for the modified E-N loading ratio.

With the Ball 3X in service, and using square pile geometries:

$$B_3 = 100$$

$$B_1 = -8.5$$

$$B_2 \text{ allowed} = 110 + \text{Ball 3X strength} = 250 \mu\text{b}$$

$$f = .680$$

$$\text{Number of E allowed} = 88 \text{ E columns}$$

Approximately thirty splines will be necessary to satisfy total control requirements during ball 3X lockout with VSR's in service.

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^{/7/} HW-69566, ibid
^{/8/} Personal Notes, W. S. Nechodom

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Figure I

Figure II

E-N FUEL LOADING AT H REACTOR

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RELATIVE NUMBER OF POISON COLUMNS VS BUCKLING EFFECT

ADDENDUM

METHOD OF ESTIMATING THE EFFECT OF POISON CHARGES

The reduction in lattice material buckling within a region associated with poison columns charged in a specific density (R_c), (regardless of the geometrical arrangement of the poisoned region) can be calculated from the formula:

$$\delta B_c = \delta B_p R_p^2 \frac{J_1^2(2.405)}{R_c^2 \left[J_0^2\left(\frac{2.405 R_c}{R_p}\right) + J_1^2\left(\frac{2.405 R_c}{R_p}\right) \right]}$$

where δB_c is the reduction in buckling
 δB_p = nominal buckling effect of the poison
 R_p = radius of the pile
 R_c = radius specifying the density of the poison in the region
 J_1 and J_0 = (Bessel Functions)

For the E-N Loading

$$\begin{aligned} \delta B_p &= .633 \mu b \text{ for poison splines} \\ &2.44 \mu b \text{ for full column poison} \\ R_p &= 600 \text{ cm} \\ R_c^2 &= \frac{1162 \times 452.5}{\#} \end{aligned}$$

where 1162 = number of tubes in flat zone
 452.5 = area per lattice (cm^2)
 # = number poison tubes with splines

The buckling reduction associated with various numbers of poison fueled tubes and splines in the H Reactor flat zone are graphed in Figure 3.

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