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CRITICALITY PREDICTIONS IN THE HANFORD REACTORS

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CRITICALITY PREDICTIONS IN THE HANFORD REACTORS

By

S. M. Skidmore
and
C. E. Bowers

March 31, 1965

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CRITICALITY PREDICTIONS IN THE HANFORD REACTORS

INTRODUCTION AND PURPOSE

The purpose of this study was to examine the general area of criticality prediction errors at the smaller ("pre-K") reactors. Such errors evolve from a number of uncertainties which are difficult if not impossible to eliminate with present measuring and accounting techniques; the magnitude of these uncertainties in aggregate was indicated in an earlier study¹ as about 1.5 milli-k, which, at the time of the study, represented an "inherent" prediction error that constituted a practical limit on accuracy. Present day computer usage has improved considerably upon this.

Predictions at the various Hanford reactors are made on the basis of a fundamental pile variable known as cold clean reactivity. This variable may be qualitatively defined as the ultimate reactivity of the room temperature reactor with all poisons and poisonous short-term fission products removed; quantitatively, it is expressed by the equation

$$CRR = R + L + WC_m - WC_g + P + \rho\tau, \quad (1)$$

where: R is reactivity in control rods,
L is reactivity in xenon,
WC_m is the so-called metal temperature reactivity effect,
WC_g is the reactivity contribution due to the heated moderator,
P is the reactivity invested in splines, poison columns, etc.,
 $\rho\tau$ is the reactivity equivalent of the reactor period.

The quantity CCR may be determined during operation by properly evaluating the temperature-reactivity coefficients C_m and C_g, or it may alternatively be determined at startup, when these effects are zero and the equation reduces to

$$CR = R + \rho\tau + L + P. \quad (2)$$

The latter equation serves as the basis for the uniform calculational approach used herein.

SUMMARY AND CONCLUSIONS

Data from 65 startups at the B, C, D, DR, and F Reactors were utilized in this study. It was found that consistent application of the uniform base reactivity approach for predicting the critical rod configuration resulted in an over-all improvement in prediction errors when compared with the techniques currently utilized by the assigned pile physicist. The average prediction error for all startups analyzed by this method was 0.98 milli-k for 38 startups; the average error reported by the physicists for the 38 startups was 1.09 milli-k. More important from a safety standpoint is the spread of errors; methods employed by the various physicists resulted in a spread of errors of -2.80 mk to 3.81 mk, while the uniform approach yielded a span of -2.48 to +2.09 c-mk. Positive errors of 2.1 mk are quite tolerable from a safety standpoint; positive errors

¹HW-60513, A Statistical Study of the Relationship Between Prediction Errors and Various Parameters, C. E. Bowers and W. L. Stiede, May 28, 1959

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of 3.8 mk, while not intolerable, are certainly uncomfortable. In fact, adoption of the uniform approach appears justifiable if it will result in no positive errors of greater than 2.1 mk, which appears to be the case.

The comparative values developed in this study are summarized in Tables I & II.

Table I

Average Prediction Errors (in c-mk) for
B, C, D, DR, and F Reactors

<u>Reactor</u>	<u>Cold Reactivity Method</u>	<u>Physicists' Method</u>
B	71/4*	129/4
C	113/10	118/10
D	111/5	93/5
DR	104/9	116/9
F	80/10	94/10
Summary:	3705/38 = 98	4145/38 = 109

*The notation is to be read as an average prediction error of 71 c-mk for four startup predictions.

Table II

Maximum Spread of Errors (in c-mk) for
B, C, D, DR, and F Reactors

<u>Reactor</u>	<u>Cold Reactivity Method</u>	<u>Physicists' Method</u>
B	-95 to +88	-163 to -38
C	-248 to +124	-20 to +253
D	-127 to +209	-20 to +253
DR	-96 to +189	-74 to +381
F	-179 to +208	-280 to +270
Summary:	-149 to +164	-128 to +245

It is concluded that this basic approach should be used in all criticality predictions, unless other considerations and techniques result in more stringent (conservative) numbers.

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DISCUSSIONPresent Approach

It is common practice among the assigned physicists to daily record the reactor variables which influence total reactivity. The key variables recorded include daily production, reactivity in rods and supplementary control, reactor atmosphere composition, and power level history; from this running tabulation it is possible to construct a reactivity balance at any time, which by definition satisfies Equation 1. The "hot operating reactivity" is thus empirically equated to the more basic "cold clean reactivity." Consistent results are obtained when pile variables are evaluated based on a known cold clean reactivity (CCR) but the converse is not true; significant errors are probable when one attempts to evaluate CCR from the mixture of calculated and observed parameters in Equation 1, and this creates the problem.

A moment's reflection will clarify the above statement. The cold reactivity is the ultimate reactivity of the unpoisoned, cold reactor; this constitutes the available reactivity which may be "invested" in the various transients which accompany operation.

By a semiempirical approach, one evaluates the basic reactivity and then systematically attempts to evaluate the transients. The difficulty which inevitably results may be traced to flux distribution changes; even small local flux variations may cause significant changes in the transient effects, all of which are power-, exposure-, and local flux-squared-dependent. This obviously is due to the fact that integrated neutron importance is the overriding factor; for example, a shift of peak flux from a region containing a high plutonium inventory to a region of low exposure may have a pile reactivity effect many times greater than fractional local power change would indicate.

Basis for Proposed Approach

The technique proposed herein does not have this disadvantage; it provides for startup predictions based wholly on observations of previous startups. Thus, one does not have to be concerned with errors in the various transients, and more important, one predicts rod configurations in a consistent frame of reference (i.e., the flux distribution in the cold pile for a particular kind of startup varies very little from one startup to the next; the same cannot be said about flux distribution changes from one shutdown to the next).

The justification for this approach can perhaps be summarized by writing Equations 1 and 2 in the more exact statistical forms:

$$(CCR)_{SD} = (R \pm \epsilon_R) + (L \pm \epsilon_L) + (WC_m \pm \epsilon_{WC_m}) - (WC_g \pm \epsilon_{WC_g}) + (P \pm \epsilon_P) + (\rho \tau \pm \epsilon_{\rho\tau}) \quad (1a)$$

$$(CCR)_{SU} = (R \pm \epsilon_R) + (L \pm \epsilon_L) + (P \pm \epsilon_P) + (\rho \tau \pm \epsilon_{\rho\tau}), \quad (2a)$$

where the ϵ_i represent uncertainties in observations. Note that two uncertainties (ϵ_{WC_m} and ϵ_{WC_g}) are eliminated by use of Equation 2a; furthermore, ϵ_R will be

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minimized since all evaluations will be in a similar flux distribution. The prediction error may be related to these uncertainties in several ways; it is reasonably accurate to do this by an equation of the form

$$P.E. = \sqrt{\epsilon_R^2 + \epsilon_L^2 + \dots + \epsilon_{PT}^2} \quad (3)$$

Even though this relationship may not be exact or complete, it is obvious that the elimination of any individual uncertainty will reduce the total uncertainty; this is borne out by the results of the present study.

"The Physicists' Method"

Each assigned physicist develops his own technique through years of experience; while the individual approaches are not different in theory, they nevertheless contain small variations peculiar to the individual calculator. Such variations might include personalized "shadowing factors" for supplementary poison, xenon calculational short cuts, modified rod calibrations, etc. Variations are obviously justified on the ground that the physicist is the single individual most cognizant of the basic reactivity of his assigned reactor. No attempt will be made to either describe or analyze any particular method used by any particular physicist; rather, documented results as tabulated in the reactor Shutdown Data Sheets will be used for comparison purposes. These documented results form the basis for the numbers in Tables 1 and 2.

The "Proposed Method" in Detail

Several items are basic in the proposed approach:

1. A xenon transient table (HW-49690 or equivalent),
2. A relationship¹¹ between basic cold reactivity and exposure (Figure I),
3. A relationship between central zone nominal reactivity and pile effect (i.e., No. Splines vs. Pile Reactivity) (Figure II),
4. A tabulation of previously observed startup reactivity (Shutdown Data Sheets),
5. An estimate of the exposure of discharged material (from Production Computing).

To calculate the critical configuration for a given startup, one first establishes the average basic reactivity for the reactor by assuming that this parameter is fairly constant. For a given pile exposure and loading, cold reactivity varies very little over reasonably long periods of time.

The constant base cold reactivity is obtained by first computing the observed reactivity for the previous one or more well defined startups. Each observed cold reactivity is corrected for differences between pile exposure and loading, with respect to the base conditions. The base conditions for pile exposure and loading are held constant at a convenient choice so the observed reactivities can be corrected to equal conditions and averaged; therefore an observed cold reactivity

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for a given set of conditions will be established. This is shown by an equation expressed as:

$$CCR_{base} = 1/n \sum_{i=1}^n [CCR_{SU} \pm \Delta R_L \pm \Delta R_E]_i \quad (4)$$

Where:

CCR_{SU} is the observed cold reactivity for a given startup.

ΔR_E is the reactivity correction to the base pile exposure, using curve 1.

ΔR_L is reactivity correction to the base pile loading

n is the number of previous startups considered in establishing basic reactivity

The resulting cold clean reactivity for the startup under consideration is now found by correcting the base cold reactivity to the pile exposure and loading at startup. This is given by:

$$CCR_{SU} = CCR_{base} \pm \Delta R'_E \pm \Delta R'_L \quad (5)$$

Where:

$\Delta R'_E$ is the reactivity correction between the base pile exposure and the startup pile exposure

$\Delta R'_L$ is the reactivity correction for the difference between the base and startup pile loading

Equation 5 provides the working relationship for predicting the critical rod configuration.

The correction terms for exposure are calculated using data from production scheduling, namely "Reactor Cover Sheets," "Reactor Discharge Summary Reports," and Figure 1. The reactivity variation due to exposure is determined by calculating the long-term gains (LTG) due to operation and the discharge loss (DL) incurred during the outage. The long-term gains correction, for CCR base, is calculated by taking the reactivity difference between the base exposure and the i th shutdown exposure. The discharge loss is calculated by flux-squared weighting the lattice reactivity loss due to replacing irradiated fuel with material at zero exposure.

The total change in pile reactivity due to exposure differences in calculating CCR_{base} is then:

$$[\Delta R_E]_i = [R_{LTG} + R_{DL}]_i \quad (6)$$

Where: $[R_{LTG}]_i = R_{E(base)} - [R_{E(SD)}]_i \quad (7)$

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$$[R_{DL}]_i = \{[f_{(disch)}][R_{E(disch)} - R_{E=C}]\} \quad (8)$$

For the exposure correction term for CCR_{SU} , one determines the reactivity difference between the base exposure and the shutdown exposure for the outage, and calculates the discharge loss for any discharge during the outage.

$$\therefore \Delta R'_E = R'_{LTG} + R'_{DL} \quad (9)$$

$$\text{Where: } R'_{LTG} = [R_{E(SD)} - R_{E(base)}] \quad (10)$$

$$R'_{DL} = \{[f^2_{(disch)}][R_{(E=0)} - R_{E(disch)}]\} \quad (11)$$

The correction terms for the differences between the pile loading and the base pile loading were calculated from the results of computer runs using the "BUCK" program.² It was found that a column of "spike" enrichment is worth 8.4 c-mk and a column of "E-Ring" enrichment is worth 7.25 c-mk with respect to natural metal. The correction equations for ΔR_L and $\Delta R'_L$ are similar in form to those for ΔR_E and $\Delta R'_E$. That is, one corrects the loading of the i th startup to the base pile loading in order to obtain $CCR_{(base)}$ and corrects the average base cold reactivity to the pile loading of the startup under consideration.

More accurate results may be obtained by using enrichment worths particular to each reactor instead of a constant worth.

Example:

I. Establish CCR_{base} :

Let the base conditions be:

$$R_{DL} = 0$$

$$R_{LTG} = 0 \quad E_{SD(base)} = 15.5 \text{ MWD/Tube}$$

$$\therefore \Delta R_{E(base)} = 0$$

Pile loading = 28 spike, 126 E-Ring

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A. Startup #1

1. $CCR_{SU} = 2338$ (observed)
2. Pile loading: 29 spike, 120 E-ring
3. $E_{SD} = 49.8$ MWD/Tube
4. $F_{disch}^2 = 0.395$
5. $E_{disch} \approx 70$ MWD/Tube

$$\begin{aligned} [R_{DL}]_1 &= [F_{disch}^2](\Delta R_{E(disch)}) \\ &= (0.395)(80) \\ &= +31 \text{ c-mk} \end{aligned}$$

$$\begin{aligned} [R_{LTG}]_1 &= R_{E(base)} - [R_{E(SD)}]_1 \\ &= 0 - 246 \\ &= -246 \text{ c-mk} \end{aligned}$$

$$\therefore [\Delta R_E]_1 = [R_{LTG} + R_{DL}]_1 = +31 - 246 = -215 \text{ c-mk}$$

$$[\Delta R_L]_1 = -8.4 + 43.5 \approx +35 \text{ c-mk}$$

$$\begin{aligned} \therefore [CCR_{base}]_1 &= [CCR_{SU} + \Delta R_E + \Delta R_L]_1 \\ &= 2338 - 215 + 35 \\ &= 2088 \text{ c-mk} \end{aligned}$$

B. Startup #2

1. $CCR_{SU} = 2333$ (observed)
2. Pile loading = 28 spike, 120 E-ring
3. $E_{SD} = 47.1$ MWD/Tube
4. $f_{disch}^2 = 0.380$
5. $E_{disch} \approx 72$ MWD/Tube

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$$[R_{DL}]_2 = (0.380)(70) = +27 \text{ c-mk}$$

$$[R_{LTG}]_2 = 0 - 235 = -235 \text{ c-mk}$$

$$\therefore [\Delta R_E]_2 = +27 - 235 = -208 \text{ c-mk}$$

$$[\Delta R_L]_2 = 0$$

$$\therefore [CCR_{base}]_2 = 2333 - 208 + 0 = 2125 \text{ c-mk}$$

C. Startup #3

$$1. CCR_{SU} = 2388$$

$$2. \text{Pile loading} = 28 \text{ spike, 127 E-ring}$$

$$3. E_{SD} = 39.1 \text{ MWD/Tube}$$

$$4. r_{disch}^2 = 0.068$$

$$5. E_{disch} \approx 73 \text{ MWD/Tube}$$

$$[R_{DL}]_3 = (0.068)(60) = 46 \text{ c-mk}$$

$$[R_{LTG}]_3 = 0 - 186 = -186 \text{ c-mk}$$

$$\therefore [\Delta R_E]_3 = +4 - 186 = -182 \text{ c-mk}$$

$$[\Delta R_L]_3 = 0$$

$$\therefore [CCR_{base}]_3 = 2388 - 182 + 0 = 2206 \text{ c-mk}$$

$$\therefore CCR_{base} = 1/3 [2088 + 2125 + 2206] \\ = 2140 \text{ c-mk}$$

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II. Calculate CCR_{SU} :

For the startup being calculated:

1. Pile loading: 30 spike, 124 E-ring
2. $E_{SD} = 59.0$ MWD/Tube
3. $r_{disch}^2 = 0.501$
4. $E_{disch} \approx 77$

$$R_{DL}' = (0.501)(165 - 232) = -34 \text{ c-mk}$$

$$R_{LTG}' = 265 + 0 = 265 \text{ c-mk}$$

$$\therefore \Delta R_E' = -34 + 265 = +231 \text{ c-mk}$$

$$\Delta R_L' = 16.8 - 14.5 \approx +2 \text{ c-mk}$$

$$\therefore CCR_{SU} = CCR_{base} + \Delta R_E' + \Delta R_L'$$

$$= 2176 + 231 + 2$$

$$= 2409 \text{ c-mk}$$

Since this calculation is fairly long, it would be useful to maintain a table of:

1. Startup date
2. R_{LTG}
3. R_{DL}
4. ΔR_L
5. $[CCR_{base}]_i$
6. CCR_{base} averaged for the last three well defined startups
7. CCR_{SU} predicted
8. Prediction error
9. Comments: which would include the amount of water collection after startup, any flux distortion at startup, etc.

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This is demonstrated by the reactor data used for this study as displayed in Table III.

Comments on Exposure Variations

Additional discussion on exposure-reactivity variations may be appropriate.

In the operating reactor, each fuel column undergoes a reactivity transient similar to that shown in Figure I. Thus the $\Delta\rho$ - ΔE relationship shown there applies to each individual lattice (e.g., $\Delta\rho_L$), and the pile effect must be obtained by suitably integrating the individual lattice effects. The flux-squared weighting technique is widely used, and usually suffices, for this integration:

$$\Delta\rho_{\text{pile}}(E) \equiv \frac{\int_V \Delta\rho_L \phi_L^2 dV}{\int_V \phi^2 dV} \sim \frac{\sum_1^N \Delta\rho_L r_L^2}{\sum_1^N r_L^2} \quad (12)$$

where the subscript L refers to the individual lattice, N refers to the total number of power producing tubes, and r^2 is the conventional "factor squared"

given by $r_i = \frac{p_i}{\bar{p}}$, with p_i denoting the i th tube power and $\bar{p}$ denoting the average tube power.

Equation 5 may be broken down into any reasonable arbitrary grouping:

$$\Delta\rho_{\text{pile}}(E) \approx \frac{\sum_1^k \Delta\rho_L r_L^2 + \sum_k^1 \Delta\rho_L r_L^2 + \dots + \sum_p^q \Delta\rho_L r_L^2 + \sum_q^N \Delta\rho_L r_L^2}{\sum_1^N r_L^2} \quad (13)$$

It appears reasonable to break the reactivity effect down into two groups, reflecting material to be discharged and material to remain:

$$\Delta\rho_{\text{pile}}(E) \approx \frac{\sum_1^n \Delta\rho_L r_L^2}{\sum_1^N r_L^2} + \frac{\sum_n^N \Delta\rho_L r_L^2}{\sum_1^N r_L^2} \quad (14)$$

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This technique is most easily applied in terms of flux-squared weighted exposure; the following approximation (although considerably less exact theoretically than Equation 7 has been found to apply with good accuracy:

$$\bar{E}_{SD} \approx \frac{\sum_{i=1}^N E_i f_i}{\sum_{i=1}^N f_i^2} \approx E_D \left(\frac{\sum_{i=1}^m f_i^2}{\sum_{i=1}^N f_i^2} \right) + E_R \left(\frac{\sum_{i=1}^N f_i^2}{\sum_{i=1}^N f_i^2} \right) \quad (15)$$

$$= 1/100 [E_D (\%r^2_{\text{discharge}}) + E_R (\%r^2_{\text{residual}})]$$

Thus, given any three of the parameters $\bar{E}_{SD}$, E_D , E_R , $\%r^2_{\text{discharge}}$, $\%r^2_{\text{residual}}$, one may calculate the others (obviously, $\%r^2_{\text{residual}} = 100 - \%r^2_{\text{discharged}}$).

When irradiated fuel is discharged, it reverts from exposure E_D to $E = 0$, and the lattice reactivity is changed correspondingly (as determined from Figure I). Thus, as shown in the stepwise procedure above, one calculates pile reactivity changes in going from $\bar{E}_0$ to $\bar{E}_1$ directly from the curve in Figure I; discharge effects are found by weighting the reactivity change in going from E_D to $E = 0$.

This same approach may be used to accurately predict the so-called "samarium dip." One first breaks the pile exposure down into two segments as shown by Equation 8; then the transient reactivity is predicted by considering the reactor as being comprised of two portions, one initially at $E = E_R$ and the other at $E = 0$.

A composite transient is then constructed for ensuing operation. For clarification, let us present a simple graphical example:

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Assume that the shutdown flux-square-weighted exposure is $\bar{E}$, consisting of some material at E_R and some at E_D (which is to be discharged); the relationship is, of course, Equation 8. Now, after discharge and subsequent startup, the residual material proceeds from exposure E_R to some higher exposure $E_R + \Delta E$, and the newly charged fuel from 0 to ΔE ; the corresponding lattice changes in cold reactivity are $\Delta\rho_2$ and $\Delta\rho_1$, respectively. The pile effect, however, is given by

$$\Delta\rho_{\text{pile}} = \frac{\%f_D^2 \Delta\rho_1 + \%f_R^2 \Delta\rho_2}{100} \quad (16)$$

with due regard to algebraic sign.

This method has been thoroughly tested and found to be quite reliable over a wide range of conditions.

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Early supporting work by James Morgan, currently with Battelle-Northwest, is gratefully acknowledged.

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Table III
Reactor Data
(Reactivity in 10^{-2} mk, or c-mk)

Base Conditions: Exposure = 15.5 MWD/Tube
 $\Delta R_{LTG} \approx 0$
 $\Delta R_{DL} \approx 0$

1. B Reactor

Base Loading: 10 spike; 127 E-ring enrichment.

SU Date	OBS CCR	ΔR_{LTG}	ΔR_{DL}	ΔR_L	Base CCR	Avg. Base CCR	Pred. CCR	CCR Error	Physicist's Error
12-11-63	2405	263	-15	-8	2165	-	-	-	-
1-10-64	2387	241	-25	+134	2037	-	-	-	-
3-13-64	2317	230	-3	+218	1872	-	-	-	-
5-6-64	2479	209	0	+340	1930	2025	2574	-95	-163
5-16-65	2444	249	-29	+190	2034	1946	2356	+88	-68
5-20-64	2215	70	-3	+158	1990	1945	2170	+45	-38
6-19-64	2298	250	-30	+149	1929	1985	2354	-56	-245

2. C Reactor

Base Loading:

SU Date	OBS CCR	ΔR_{LTG}	ΔR_{DL}	ΔR_L	Base CCR	Avg. Base CCR	Pred. CCR	CCR Error	Physicist's Error
12-6-63	2566	250	+2	-1	2315	-	-	-	-
12-19-63	2388	263	-9	+156	1978	-	-	-	-
12-25-63	2334	121	0	+156	2057	-	-	-	-
1-12-64	2571	244	-6	+92	2241	2117	2447	+124	+31
1-22-64	2488	223	-27	+239	2053	2092	2527	-39	-33
2-11-64	2687	189	-6	+276	2228	2117	2576	+111	+300
2-29-64	2476	257	-24	+289	1954	2174	2696	-220	+17
3-6-64	2430	105	0	+289	2120	2078	2472	+42	+51
3-27-64	2662	257	-25	+499	1933	2101	2830	-168	+138
4-15-64	2807	244	-4	+489	2078	2002	2731	+76	+360
5-2-64	2652	255	-33	+634	1796	2044	2900	-248	+132
5-5-64	2618	61	0	+634	1923	1936	2631	-13	-100
6-15-64	2566	93	0	+632	1841	1932	2657	-91	-21

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3. D Reactor

Base Loading: 10 spike; 125 E-ring enrichment

SU Date	OBS CCR	ΔR_{LTG}	ΔR_{DL}	ΔR_L	Base CCR	Avg. Base CCR	Pred. CCR	CCR Error	Physicist's Error
10-29-63	1887	159	-1	+269	1460	-	-	-	-
11-5-63	1961	199	+8	+326	1428	-	-	-	-
12-14-63	2042	244	0	+61	1737	-	-	-	-
1-16-64	2333	209	+40	+333	1751	1542	2124	+209	+112
2-24-64	2538	256	-12	+593	1701	1639	2476	+62	+76
3-9-64	2347	197	-7	+509	1603	1730	2429	-127	-5
4-16-64	2302	264	-37	+419	1656	1685	2331	-29	-20
4-24-64	2299	37	-1	+482	1781	1653	2171	+128	+253

4. DR Reactor

Base Loading: 47 spike; 147 E-ring enrichment

SU Date	OBS CCR	ΔR_{LTG}	ΔR_{DL}	ΔR_L	Base CCR	Avg. Base CCR	Pred. CCR	CCR Error	Physicist's Error
10-9-63	1926	18	0	+186	1722	-	-	-	-
11-28-63	1760	138	0	0	1622	-	-	-	-
12-31-63	2113	72	-8	+275	1774	-	-	-	-
1-7-64	1938	53	0	+275	1610	1706	2034	-96	-44
1-12-64	1946	84	0	+275	1587	1669	2028	-82	-25
2-3-64	2273	248	-26	+275	1776	1657	2154	+119	+190
3-11-64	2328	264	-15	+275	1804	1658	2182	+146	+165
4-22-64	2287	261	-26	+141	1911	1722	2098	+189	+381
5-3-64	2144	54	-3	+94	1999	1830	1975	+169	-20
5-7-64	2047	58	-1	+72	1918	1905	2034	+13	0
5-17-64	2199	145	-3	+72	1902	1943	2158	-41	-75
7-2-64	2114	252	0	0	1862	1940	2192	-78	+100

5. F Reactor

Base Loading: 14 spike; 139 E-ring enrichment

SU Date	OBS CCR	ΔR_{LTG}	ΔR_{DL}	ΔR_L	Base CCR	Avg. Base CCR	Pred. CCR	CCR Error	Physicist's Error
12-11-63	2010	72	0	0	1938	-	-	-	-
12-23-63	2100	162	0	+168	1770	-	-	-	-
1-29-64	2200	257	-48	+114	1877	-	-	-	-
2-8-64	1970	96	0	+21	1853	1862	1979	-9	-280
2-26-64	2072	229	-2	+31	1814	1833	2091	-19	+35
3-15-64	2120	259	-44	+106	1799	1848	2169	-49	+25
4-2-64	2091	165	0	+106	1820	1822	2093	-2	+30
4-19-64	2200	249	-40	+71	1920	1811	2091	+109	-50
4-24-64	2170	45	0	+121	2004	1846	2012	+158	+10
5-4-64	2185	135	0	+121	1901	1915	2171	+14	-115
5-6-64	2444	135	-4	+163	2150	1942	2236	+208	+270
6-6-64	2040	111	-1	+91	1839	2018	2219	-179	-110
7-1-64	2295	265	-34	+158	1906	1963	2352	-57	-10

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Table IV

Regular Spline Reactivity Worths

<u>Feet in Pile</u>	<u>Reactivity Loss, c-mk</u>
12	0
13	1
14	1
15	1
16	2
17	3
18	4
19	5
20	7
21	8
22	10
23	11
24	13
25	14
26	16
27	17
28	18
29	18
30	19
31	19
32	20

All partial splines should be converted to equivalent full length splines before using Figure 2.

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1. R_N (Spline) = 45 c-mk
2. R_N (Pb-Cd) = 160 c-mk
3. Add all nominal worths of poisons together before finding the effective worth.

Figure 2

Nominal vs. Effective Reactivity Worths

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END