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HANFORD TECHNICAL RECORD

HW-75904

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HANFORD ATOMIC PRODUCTS OPERATION - RICHLAND, WASHINGTON

December 14, 1962

TITLE

ENT'D

REACTIVITY ACCOUNTING BY "CALHOT", A ONE-DIMENSION BUCKLING CODE

AUTHOR

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R. T. Toyooka

OTHER OFFICIAL CLASSIFIED INFORMATION

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REACTIVITY ACCOUNTING BY "CALHOT", A ONE-GROUP
ONE-DIMENSION BUCKLING CODE

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REACTIVITY ACCOUNTING BY "CALHOT", A ONE-GROUP
ONE-DIMENSION BUCKLING CODE

INTRODUCTION AND PURPOSE

Normal pile exposure variations and day-to-day operating reactivity adjustments must be compensated by a combination of in-reactor poison and enrichment. The calculation of current reactivity status and the prediction of future trends is an integral part of the Physicist's responsibility. This document describes the computer program "CALHOT", which is based upon one-group diffusion theory^{/1/}, and is intended to be used in routine reactivity-enrichment calculations. It offers the advantages over present hand calculations of greater accuracy, more flexibility, and, most important, much less calculational time. Basic familiarization with Reference 1 is desirable for optimum utilization of this program.

SUMMARY

"CALHOT" may be used to perform routine reactivity and flattening calculations for multiregion (maximum 15) Hanford production reactors. The main body of this report briefly describes the general problem for which CALHOT provides a solution, and also includes complete input and output descriptions. The code will, for proper input instructions ("ICALC"), calculate either individual regional buckling changes required for criticality (local effects) or uniform pile buckling changes (pile reactivity).

The Appendix provides a rather detailed discussion of the basic assumptions and calculational techniques utilized, with a complete FORTRAN program listing. It is included for possible future reference, to provide background information in the event program changes are appropriate at some later date; also included is a sample calculational run, including input and output.

Program input includes the standard reactor geometry, regional bucklings, and boundary conditions (standardized temperature and exposure transient effects and fission product poisoning calculational techniques are "built in"). Corrections in regional bucklings for nonuranium material, control rod insertion, etc., are to be made by the user.

Program output includes hot and cold pile reactivity, enrichment (both "spike" and "ring") worth, and control rod strength.

It is felt that the program in its present form is adequate to the task of computing routine "spike" enrichment and "poison" requirements to maintain a given degree of flattening. It is expected that many more specialized uses will be found for the program as experience is accumulated.

BRIEF DESCRIPTION OF THE PROBLEM

In reactor calculations it is the general practice to establish the condition of the reactor when just critical. A general equation representing the neutron balance at critical can be written as -

/1/ HW-70780 - An Operational Reactivity Accounting System Based on One-Group
Diffusion Theory, G.F. Bailey, April, 1962 (Secret)

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Production - Leakage - Absorption = 0

or

$$S + D\nabla^2\phi = \Sigma_a\phi = 0$$

which can also be written

$$\nabla^2\phi + B^2\phi = 0$$

assuming the source term proportional to the flux.

In applying this buckling equation it is necessary to approximate the complex reactor shape with a simple, convenient geometric figure, generally an infinite slab or cylinder.

The Laplacian operator is then expressed by the proper geometric functions, the appropriate boundary conditions applied and the wave equation is solved for the lowest eigen value that will give a solution. A complete derivation and explanation of the diffusion equation and its reactor applications is given in Reference 2.

The existing Hanford reactors are approximated by assuming their shape is a right cylinder and the regions with marked differences in fuel loading are assumed to be individual concentric rings within the right cylinder. The regional buckling is expressed as follows:

$$B_I^2 = B_M^2 - B_G^2 + \delta B_X^2 + \delta B_S^2 + \delta B_N^2 + \delta B_{LTG}^2 + \delta B_M^2 + \delta B_{MOD}^2 + \delta B_{con}^2 + \delta B_{rod}^2 + \delta B_{hur}^2$$

where:

B_I^2 = the effective regional buckling

B_M^2 = three dimensional cold lattice material buckling

B_G^2 = axial geometric buckling

δB_X^2 = change in buckling due to xenon

δB_S^2 = change in buckling due to samarium

δB_N^2 = change in buckling due to neptunium

δB_{LTG}^2 = change in buckling due to cold long-term gains

δB_M^2 = change in buckling due to metal heating

δB_{MOD}^2 = change in buckling due to graphite heating, a first approximation

If this program is to be used effectively for predicting enrichment requirements, the reactor physicist should have available a curve of maximum allowable δB_{MOD}^2 versus exposure for the present graphite limits. If the δB_{MOD}^2 required for the center region is higher than allowed by the curve, the difference must be added as spike enrichment.

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δB_{con}^2
 δB_{rod}^2
 δB_{nur}^2 } change in buckling due to rods, supplementary control or inactive tubes.

INPUT LISTING AND EXPLANATION

INPUT CARD ONE

- This card (the "Date" card) is the first card following the data card and only one is required for any machine run.

DATE

(Columns 1-24) - Date of machine run.

IBOUND

(Columns 25-30) - Iteration limit on convergence routine for the critical calculation. Maximum is 40, normal problem requires less than 8.

PHITOL

(Columns 31-40) - Tolerance limits on flux. For example 0.00001

UTOL

(Columns 41-50) - Tolerance limits on critical buckling. For example 0.01 would be $\pm 0.01 \mu b$.

CASE CARD ONE

- (5A6, I6, 6 PF 10.5 - ZPF 10.5, 2 PF 8.3)

CASE

(Columns 1-30) - Case identification

INPUT

(Columns 31-36) - Buckling option
If input = 1, first ΔB^2 guess calculated by the program.
If input = 2, unknown bucklings are inputs in ΔB^2

DELT B^2

(Columns 37-46) - Uniform Delta buckling to be added to the first buckling guess (when Input = 2); subsequent trials T_1 will lie within the interval
(First guess $+0.25 \Delta B^2$) $\leq T_1 \leq$ (first guess $+ \Delta B^2$)

PWR

(Columns 47-56) - Pile power multiplied by 10^{-2}

To(1)

(Columns 57-64) - Length of last outage in days

To(Z)

(Columns 65-72) - Length of time in days since last outage. If it is 8 days or less, transient effects of N_p and P_m are calculated.

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CASE CARD TWO (12I6)

NR
(Columns 1-6) - Number of regions in the problem (maximum 15)

IB1
(Columns 7-12) - First region of unknown buckling ($1 \leq IB1 \leq IB2$)

IB2
(Columns 13-18) - Last region of unknown buckling ($IB1 \leq IB2 \leq NR$)

IFC
(Columns 19-24) - Final conditions specification:
IFC = 1 zero flux at outer boundary
IFC = 2 zero current at outer boundary

ICALC
(Columns 25-30) - Calculation Code:
ICALC = 1, Delta B^2 is applied from IB1 through IB2, including IB1.
ICALC = 2, Delta B^2 is applied in IB1 and IB2 only.
ICALC = 3, Delta B^2 is applied from 1 through IB1, and from IB2 through NR respectively, including 1 and IB2.

ICONO
(Columns 31-36) - Initial region in which control worth is to be considered.
(ICONO $\leq$ ICONL)

ICONL
(Columns 37-42) - Final region in which control worth is to be considered.

INATO
(Columns 43-48) - Initial region of center to find spike worth (INATO $\leq$ INATL)

INATL
(Columns 49-54) - Final region of the center, spike containing portion of the reactor.

IENRO
(Columns 55-60) - Initial region of the enrichment ring (IENRO $\leq$ IENRL)

ENRL
(Columns 61-66) - Final region of the enrichment ring.

IREACT
(Columns 67-72) - Reactor Code:
IREACT = -1: B,D,DR and F Reactors
IREACT = 0: C and H Reactors
IREACT = 1: KE and KW Reactors

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CASE CARD THREE (OPF 12.5, 6P4F 12.5)

RREAD

(Columns 1-12) - Region radius from origin in centimeters

BLEAK

(Columns 13-24) - Axial geometric buckling in μb :

11.72 μb for K reactors

13.61 μb for old reactors with 34-piece columns

15.17 μb for old reactors with 32-piece columns

BCON

(Columns 25-36) - Local buckling effect of supplementary poison in μb .

BROD

(Columns 37-48) - Local buckling effect of control rods in μb .

BNUR

(Columns 49-60) - Local buckling effect of non-uranium or inactive tubes in μb .

A separate "CASE CARD THREE" is required for every region; the same is true for "CASE CARD FOUR." Thus for n regions, there will be n "CASE CARDS THREE" followed by n "CASE CARDS FOUR."

CASE CARD FOUR (6F 12.4)

AVZFAC

(Columns 1-12) - Average region factor

EP (I,1)

(Columns 13-24) - Average exposure of natural fuel (in MWD/T)

EP (I,2)

(Columns 25-36) - Average exposure of enriched fuel

NAT

(Columns 37-48) - Fraction of natural fuel in the region (for example, 0.92 for 92% natural uranium)

ENR

(Columns 49-60) - Fraction of enriched fuel

DISCH

(Columns 61-72) - Fraction of F^2 of the discharge

An example of the input format follows; appropriate decimal point locations are indicated. Also included is a copy of an actual machine run.

80 - 80																																								REQUESTED BY										DATE										PROBLEM NO.																																																																					
																																								XXXX										XXXX										CUS XXXX																																																																					
1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65	66	67	68	69	70	71	72	73	74	75	76	77	78	79	80																																																		
DATE										IBOUND										PHITOL										UTOL										DATE																																																																																									
CASE										INPUT										DELT B2										PWR										TO(1)										TO(2)										CASE																																																																					
NR										IB1										IB2										IFC										ICALO										ICONO										ICONL										INATO										INATL										IENRO										IENR										LIREACT										NR									
RREAD										BLEAK										BCON										BR										OD										BNUR										RAT										ENR										DISCH										RREAD																																							
AVZFAC										EP(1)										EP(2)										RAT										ENR										DISCH										AVZFAC																																																																					
Card 3 continues with a maximum of fifteen cards - each card represents a region.																																																																																																																																	
Card 4 continues the same as card 3.																																																																																																																																	
*Note: This card is only used once; subsequent cases do not require it.																																																																																																																																	
**This card is "Card One" for each case, the next is "Card Two," etc.																																																																																																																																	

INPUT TO CALHOT

NR 3	IB1 1	IB2 3	IFC 1	ICALC 1	ICONO 1	ICONL 1	INATO 1	INATL 1	IENRO 2	IENRL 2	IREACT 1	IBOUND 40							
<table border="0"> <tr> <td>INPUT 1</td> <td>PHITOL 0.00001000</td> <td>UTOL 0.10000</td> <td>DELTB2 -0.</td> <td>W 44.000</td> <td>TOX1 -0.</td> <td>TOX2 8.500</td> </tr> </table>													INPUT 1	PHITOL 0.00001000	UTOL 0.10000	DELTB2 -0.	W 44.000	TOX1 -0.	TOX2 8.500
INPUT 1	PHITOL 0.00001000	UTOL 0.10000	DELTB2 -0.	W 44.000	TOX1 -0.	TOX2 8.500													
RREAD 459.1000 504.6000 650.6000	BLEAK 11.7200 11.7200 11.7200	BCON 8.4000 -0. -0.	BROD 16.1000 -0. -0.	BNUR 3.0900 2.2000 2.2000	AVZFAC 1.136 1.027 0.656	EP-NAT 436.000 568.000 426.000	EP-ENR 310.000 393.000 0.	NAT 0.875 0.251 1.000	ENR 0.125 0.749 0.	DISCH -0. -0. -0.									

HOT BUCKLINGS

CASE #KW REACTOR 2-8-63

DATE #FEBRUARY 8 1963

RUN # 1

REGION	MATERIAL	XENON	SAMARIUM	NEPTUNIUM	L.T.G.	MODERATOR	METAL	LEAKAGE	CONTROL	ROD	NON U
1	79.1625	-51.7210	-10.0587	-8.9473	10.9289	34.3173	-14.3716	-11.7200	-8.4000	-16.1000	-3.0900
2	215.2569	-51.7061	-10.3520	-7.0971	4.3192	30.8051	-14.6902	-11.7200	0.	0.	-2.2000
3	51.9000	-50.2041	-10.0000	-5.2937	12.1723	20.5461	-8.0819	-11.7200	0.	0.	-2.2000

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****CALHOT OUTPUT****

CASE #KW REACTOR 2-8-63

DATE #FEBRUARY 8 1963

RUN # 1

REGION	DISTANCE	BREADTH	BUCKLING	AXIS	CXIS	FLUX	SLOPE	LEAKAGE
1	459.1000	-0.0000	1.3380	1.00E 00	0.	0.9307	-0.0003	0.0002
2	504.6000	152.6158	153.9538	6.60E-01	-2.72E 00	0.7786	-0.0061	0.0048
3	650.6000	-2.8812	-1.5432	-1.70E 00	3.57E 00	0.0000	-0.0047	0.0036

SOLUTION REQUIRED 5 ITERATIONS. LAST FIVE VALUES WERE

0.
-0.9992
0.6831
1.5174
0.1283

BG # 1.36625E-05 SUMA # 1.32977E 06 SUMB # 1.94967E 01 AREAU # 1.32977E 06

DECLASSIFIED

WORTH OF CONTROLS

CASE #KW REACTOR 2-8-63

DATE #FEBRUARY 8 1963

RUN # 1

REGION	MATERIAL	XENON	SAMARIUM	NEPTUNIUM	L.T.G.	MODERATOR	METAL	LEAKAGE	CONTROL	NON U
1	79.1625	-51.7210	-10.0587	-8.9473	10.9289	34.3173	-14.3716	-11.7200	0.	-3.0900
2	215.2569	-51.7061	-10.3520	-7.0971	4.3192	30.8051	-14.6902	-11.7200	0.	-2.2000
3	51.9000	-50.2041	-10.0000	-5.2937	12.1723	20.5461	-8.0819	-11.7200	0.	-2.2000

DECLASSIFIED

CALHOT OUTPUT

CASE #KW REACTOR 2-8-63

DATE #FEBRUARY 8 1963

RUN # 1

REGION	DISTANCE	BREADTH	BUCKLING	AXIS	CZIS	FLUX	SLOPE	LEAKAGE
1	459.1000	24.5000	6.4052	1.00E 00	0.	0.6899	-0.0012	0.0010
2	504.6000	152.6158	134.5209	-5.09E-01	-1.94E 00	0.5478	-0.0048	0.0037
3	650.6000	-2.8812	-20.9761	-7.12E-02	9.62E 00	0.0000	-0.0031	0.0024

SOLUTION REQUIRED 6 ITERATIONS. LAST FIVE VALUES WERE

-13.1990
-0.6831
-5.2314
1.1964
-0.1709

BG # 1.36625E-05 SUMA # 1.32977E 06 SUMB # 3.57197E 01 AREAU # 1.32977E 06

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CENTER ENRICHMENT WORTH

CASE #KW REACTOR 2-8-63

DATE #FEBRUARY 8 1963

RUN # 1

REGION	MATERIAL	XENON	SAMARIUM	NEPTUNIUM	L.T.G.	MODERATOR	METAL	LEAKAGE	CONTROL	ROD	NON U
1	51.9000	-51.7210	-10.0587	-8.9473	10.9289	34.3173	-14.3716	-11.7200	-8.4000	-16.1000	-3.0900
2	215.2569	-51.7061	-10.3520	-7.0971	4.3192	30.8051	-14.6902	-11.7200	0.	0.	-2.2000
3	51.9000	-50.2041	-10.0000	-5.2937	12.1723	20.5461	-8.0819	-11.7200	0.	0.	-2.2000

RECEIVED

****CALHOT OUTPUT****

CASE #KW REACTOR 2-8-63

DATE #FEBRUARY 8 1963

RUN # 1

REGION	DISTANCE	BREADTH	BUCKLING	AZI	CZI	FLUX	SLOPE	LEAKAGE
1	459.1000	-27.2625	-6.6367	1.00E 00	0.	1.3815	0.0018	-0.0014
2	504.6000	152.6158	173.2416	2.80E 00	-3.31E 00	1.2222	-0.0084	0.0066
3	650.6000	-2.8812	17.7446	3.63E 00	1.29E 00	0.0000	-0.0079	0.0061

SOLUTION REQUIRED 7 ITERATIONS. LAST FIVE VALUES WERE

0.6831
5.7156
1.2703
0.3576
0.0226

BG # 1.36625E-05 SUMA # 1.32977E 06 SUMB # 1.44453E 00 AREAU # 1.32977E 06

ENRICHMENT RING WORTH

CASE #KW REACTOR 2-8-63

DATE #FEBRUARY 8 1963

RUN # 1

REGION	MATERIAL	XENON	SAMARIUM	NEPTUNIUM	L.T.G.	MODERATOR	METAL	LEAKAGE	CONTROL	ROD	NON U
1	79.1625	-51.7210	-10.0587	-8.9473	10.9289	34.3173	-14.3716	-11.7200	-8.4000	-16.1000	-3.0900
2	51.9000	-51.7061	-10.3520	-7.0971	4.3192	30.8051	-14.6902	-11.7200	0.	0.	-2.2000
3	51.9000	-50.2041	-10.0000	-5.2937	12.1723	20.5461	-8.0819	-11.7200	0.	0.	-2.2000

****CALHOT OUTPUT****

CASE #KW REACTOR 2-8-63

DATE #FEBRUARY 8 1963

RUN # 1

REGION	DISTANCE	BREADTH	BUCKLING	AZIM	CZIM	FLUX	SLOPE	LEAKAGE
1	459.1000	-0.0000	14.2631	1.00E 00	0.	0.3784	-0.0022	0.0017
2	504.6000	-10.7411	3.5220	4.22E-01	-1.10E 00	0.2822	-0.0020	0.0016
3	650.6000	-2.8812	11.3819	9.46E-01	-2.06E-01	-0.0000	-0.0018	0.0014

SOLUTION REQUIRED 5 ITERATIONS. LAST FIVE VALUES WERE

0.
15.9232
-0.6831
-1.0394
0.0647

BG # 1.36625E-05 SUMA # 1.32977E 06 SUMB # -3.00628E 00 AREAU # 1.32977E 06

COLD CLEAN

CASE #KW REACTOR 2-8-63

DATE #FEBRUARY 8 1963

RUN # 1

REGION	MATERIAL	XENON	SAMARIUM	NEPTUNIUM	L.T.G.	MODERATOR	METAL	LEAKAGE	CONTROL	ROD	NON U
1	79.1625	0.	0.	0.	0.	0.	0.	-11.7200	0.	0.	0.
2	215.2569	0.	0.	0.	0.	0.	0.	-11.7200	0.	0.	0.
	51.9000	0.	0.	0.	0.	0.	0.	-11.7200	0.	0.	0.

****CALHOT OUTPUT****

CASE #KW REACTOR 2-8-63

DATE #FEBRUARY 8 1963

RUN # 1

REGION	DISTANCE	BREADTH	BUCKLING	AZI	CZI	FLUX	SLOPE	LEAKAGE
1	459.1000	67.4425	5.6048	1.00E 00	0.	0.7258	-0.0011	0.0009
2	504.6000	203.5369	141.6992	-1.97E-01	-2.12E 00	0.5795	-0.0051	0.0040
3	650.6000	40.1800	-21.6577	-7.11E-02	1.06E 01	0.0000	-0.0032	0.0025

SOLUTION REQUIRED 6 ITERATIONS. LAST FIVE VALUES WERE

-57.0153
-0.6831
-5.1236
1.1534
-0.1627

BG # 1.36625E-05 SUMA # 1.32977E 06 SUMB # 9.39855E 01 AREAU # 1.32977E 06

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OUTPUT EXPLANATION

Five critical calculations are made by the "CALHOT" program. They are:

1. The Hot Critical Buckling - where all the buckling effects are listed for each region. In addition, the following quantities are printed as output:

DISTANCE - Region radius

HREAD - First approximation of the critical buckling

BUCKLING - The critical buckling in microbucks

A(1) & C(1) - Coefficients for the flux equation

FLUX - Relative flux at boundary

SLOPE - Flux, slope at boundary

LEAKAGE - Diffusion coefficient x slope
2. Worth of Controls - where the uniform buckling change over the reactor required to compensate and the δB^2 of the controls (both rods and supplementary poison) is found.
3. Central Enrichment Worth - where the material buckling of the center is reduced to what it would be without enrichment (i.e., all enriched replaced with natural) and the uniform buckling change required for critical is found. This is by definition the worth of the spike enrichment.
4. Enrichment Ring Worth - where the material buckling of the enrichment ring is reduced to what it would be without the enrichment and the uniform buckling change required for critical is calculated. The net change is equal to the enrichment ring worth.
5. Cold Clean - in which the hypothetical reactivity of the reactor loaded with clean green fuel is calculated. By calculating the uniform buckling change required for critical with any given loading the effective reactivity worth of that particular fuel loading can be determined.

Ronald A. Chitwood
Process Physics Studies
Operational Physics Sub-Section
Research and Engineering Section

RA Chitwood:RT Toyooka:gs

Appendix

COMPONENT BUCKLING CALCULATIONS

A. Material Buckling Calculations

The following material buckling equation is used for all reactors, in all defined regions.

$$B_{MAT}^2 = \left[f(I\&E) B_{I\&E}^2 + (1-f)(I\&E^2) B_{I\&E^2}^2 \right] \mu b \quad (1)$$

where f is the fractional part of the fuel that is natural uranium, and $B_{(i)}^2$ is the wet material buckling that is given in Table 1 (the subscript refers to either natural fuel I&E or enriched fuel I&E²).

Table 1 lists the wet material bucklings presently assumed.

Table 1

Wet Material Buckling

	<u>B, D, DR, F,</u>	<u>C, H</u>	<u>KE, KW</u>
I&E	60.1	47.2	51.9
I&E ²	270	265	270

B. Xenon Calculations

Because of its high absorption cross section for thermal neutrons, xenon-135 is a fission product of particular interest. Xenon is formed mainly as a decay product of tellurium (formed in about 5.6 per cent of all fissions) which undergoes a series of beta decays to form xenon as an intermediate step.^{/3/}

The following equations are used by the program to approximate the equilibrium concentration of xenon in a Hanford reactor.^{/3,4/}

$$(\delta B_X^2)_K = \frac{(1.1658 \times 10^{-1}) W^{**}}{1 + (1.976 \times 10^{-3}) W^{**}} \mu b \quad (2)$$

$$(\delta B_X^2)_{Old} = \frac{(1.9276 \times 10^{-1}) W^{**}}{1 + (3.823 \times 10^{-3}) W^{**}} \mu b$$

/3/ Glasstone, S., and M.C. Edlund - The Elements of Nuclear Reactor Theory, New York: Van Nostrand 1952

/4/ HW-60810, Xenon Calculations for I&E Fuel Elements, R.A. Chitwood, 7-2-59

/5/ Personal Communication - D.W. Constable, April, 1963

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Where: W^{**} = (Pile Power)(Average Factor) ($\overline{df}$)($\overline{mf}$)
 Pile Power = Power in Megawatts
 Average Factor = Average region factor (available from
 production scheduling
 $= \sum_{i=1}^n f_i / n$ (where n is number of tubes in region of
 interest,
 and $f_i = \frac{(\text{Tube Power})(\text{No. Tubes in Pile})}{\text{Pile Power (KW)}}$

$$\overline{df} = 1.388 - \sqrt{.59 (\text{average factor})}$$

This is a flux correction factor arising from the fact that xenon poisoning in a flattened pile at a given power level is lower than in a non-flattened one.

($\overline{mf}$) A material correction factor for the I&E enriched fuel (the xenon buildup is slightly different in enriched than in natural I&E fuel, and the power generation is greater in a given flux).

$$(\overline{mf}) = f(\text{I\&E}) + f(\text{I\&E}^2)(X_{E2c})(P_{E2c})$$

Table 2

<u>Symbol</u>	<u>Old Reactors</u>	<u>K Reactors</u>
X_{E2c}	0.79	0.84
P_{E2c}	1.215	1.215

X_{E2c} is the enriched fuel xenon correction factor.

P_{E2c} is the enriched fuel power correction factor.

C. Neptunium Calculations /6/

The calculation of cold long term gains is often based upon the assumption of instantaneous transmutation from uranium to plutonium; this is not actually the case, since U^{238} undergoes an intermediate step during its transmutation to Pu^{239} in which Np^{239} is formed:

Thus the reactivity status should be corrected by the amount of potential plutonium that is held up in the form of neptunium.

As in reference 1, the neptunium effect is calculated by the program using the following equation:

$$N_P = N_P^0 e^{-\lambda_N t} + \frac{\bar{\Sigma}_a \phi}{\lambda_N} (1 - e^{-\lambda_N t}) \quad (3)$$

where N_P is the neptunium concentration, nuclei/cm³
 N_P^0 is neptunium at startup
 $\bar{\Sigma}_a$ is averaged lattice absorption cross section
 ϕ is reactor flux
 λ_N is neptunium decay constant
 t is time operated

Note that the saturation effect of neptunium is proportional to $\frac{\bar{\Sigma}_a \phi}{\lambda_N}$, and that, for zero power, the neptunium transient is given by $N_P^0 e^{-\lambda_N t}$

The saturation neptunium effect is corrected for reactor flattening efficiency and fuel type and converted to a local buckling effect as per the following equations ($T_0(Z) > 8$):

$$\begin{aligned} \text{(K Reactors)} \quad \delta_{BN} \text{ (Equilibrium)} &= \left\{ \left[18.3 \times 10^{-4} f + 12.2 \times 10^{-4} (1-f) (1.215)^{W*} \right] \mu_b \right\} \\ \text{(Old Reactors)} \quad \delta_{BN} \text{ (Equilibrium)} &= \left\{ \left[26.7 \times 10^{-4} f + 18.5 \times 10^{-4} (1-f) (1.215)^{W*} \right] \mu_b \right\} \end{aligned} \quad (4)$$

where f is fraction of fuel charged natural uranium
 W^* is (Pile Power)(Average Region Factor)

Neptunium holdup is calculated in the nonequilibrium case ($T_0(Z) \leq 8$) from a relationship developed by rewriting equation (3) in terms of local buckling effect:

$$\begin{aligned} \delta_{BN}(t) &= \left[\delta_{BN}^0 e^{-\lambda_N t} + \delta_{BN}^E(P) (1 - e^{-\lambda_N t}) \right] \mu_b \quad (3') \\ &= \left[\delta_{BN}^E(P) + (\delta_{BN}^0 - \delta_{BN}^E(P)) e^{-\lambda_N t} \right] \mu_b \quad (3'') \end{aligned}$$

where $\delta_{BN}(t)$ is local buckling at time t after startup
 δ_{BN}^0 is initial poisoning effect at $t = 0$
 $\delta_{BN}^E(P)$ is equilibrium neptunium effect for power level P

Note that for a shutdown case, the program makes two passes through equation (3'). First, equation (3') must be used to evaluate δ_{BN} , the value at shutdown; next, another pass through (3') is made, this time using only the second term, $\delta_{BN}^E(P) e^{-\lambda_N t}$ (t in this case is time down).

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If a discharge is taken during the outage, δ_{BN} must be reduced by the " $\%f^2$ " of the discharge ($\%f^2 = (100) \frac{\sum f^2}{N}$ where n is number of tubes discharged and N is number of active tubes).

D. Samarium Calculations

Samarium 149, a stable isotope with high cross section, is the product of beta decay of promethium 149. The equilibrium samarium poisoning effect is only about one-fifth as much that of xenon, but it is significant in that it is a stable product and doesn't decay during shutdown. The equilibrium poisoning effect in microbucks is found using the following equations ($T_0(Z) \geq 8$):

K Piles

$$\delta_{BS}^2 = S_m = \left[f(I\&E) 10.00 + (1-f)(I\&E^2) 10.47 \right] \mu b$$

Old Piles

$$\delta_{BS}^2 = S_m = \left[f(I\&E) 8.79 + (1-f)(I\&E^2) 8.90 \right] \mu b$$

(5)

Nonequilibrium samarium poisoning (for less than eight days' operation $T_0(Z) < 8$) is calculated from the basic buildup-decay relationship:

$$\delta_{BS}^2(\text{nonequilibrium}) = S_m^t = \left\{ S_m^E + P_f (P_m^E - P_m^O) e^{-\lambda_P t} - \left[S_m^E - S_m^O + P_f (P_m^E - P_m^O) \right] e^{-\sigma_B \phi t} \right\} \mu b \quad (5a)$$

S_m^t = the poisoning effect (in μb) of samarium in the reactor at time t

S_m^O = the initial condition; poisoning effect at $t=0$

S_m^E = the equilibrium samarium effect (either operating or shutdown)

E. Cold Long Term Gains

The basic cold reactivity of a reactor changes with fuel exposure.

The program assumes an exposure dependence of the form:

$$\delta R_{LTG} = (a_1 E + a_2 E^2 + a_3 E^3 + a_4 E^4) \mu b \quad (6)$$

where E is the flux-squared weighted exposure in MWD/T, and the a_i are appropriate constants as given in the following table:

These constants will generate the long term gains curves in Reference 7.

Table 3

K Reactor		Old Reactor	
I&E	I&E ²	I&E	I&E ²
a ₁ 0.0473	0.0091	0.0333	0.0020
a ₂ -0.0534 x 10 ⁻³	-0.0182 x 10 ⁻³	-0.0443 x 10 ⁻³	-0.0186 x 10 ⁻³
a ₃ 0.0235 x 10 ⁻⁶	0.0077 x 10 ⁻⁶	0.0192 x 10 ⁻⁶	0.0063 x 10 ⁻⁶
a ₄ -0.0314 x 10 ⁻¹⁰	-0.0124 x 10 ⁻¹⁰	-0.0372 x 10 ⁻¹⁰	-0.0107 x 10 ⁻¹⁰

Now, using the corresponding values in Table 3 for equation (6), the program calculates the full pile effect as follows:

$$\delta B_{LTG}^2 = [f \delta R_{LTG_{I\&E}} + (1-f) \delta R_{LTG_{I\&E^2}}] \mu b \quad (7)$$

where f is the fraction of the loading charged natural uranium (I&E).

F. Metal Effect Calculations

The effect of fuel heating is calculated by the program using the equation

$$\delta B_M^2 = (C_m W^*) \mu b$$

where C_m has the following values:

K Reactors

$$C_m = [f(I\&E) 0.0028 + (1-f) I\&E^2 (0.0028)(1.215)] \mu b/MW \quad (8)$$

Old Reactors

$$C_m = [f(I\&E) 0.0049 + (1-f) I\&E^2 (0.0049)(1.215)] \mu b/MW$$

where f is the fraction charged natural (I&E) fuel in the region.

G. Moderator Coefficient

In all of the Hanford reactors an enrichment ring is used to extend the size of the center "flattened" region. The program assumes as a first approximation that the critical B_1^2 of the flattened zone is zero; actually, the critical buckling is nearly always quite close to zero so this estimate is not bad.

The first approximation of zero buckling for the "flat" zone is necessary for assigning an initial value for δB_{MOD}^2 ; this initial value is simply the difference between zero and the summation of all central zone bucklings (assumed known) other than δB_{MOD}^2 . Assuming this trial value is correct for a base

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local power, δB_{MOD}^2 is power-corrected for the other regions, and a check is made to see if the resulting bucklings allow satisfaction of the diffusion equation with the conventional boundary conditions. Corrections are made to the regional B_i^2 on a programmed basis until a solution is found. The final value for δB_{MOD}^2 is not included as output, but can easily be found by adding the critical buckling of region one to each of the $(\delta B_{MOD}^2)_i$ listed on the hot buckling output. The general method is summarized in the following paragraphs.

The moderator coefficient in c-mk/(degree C) is calculated as follows for both the "K" and the "old" using the generalized relationship.⁷⁷

$$C_T = 2.55 + (AE + BE^2 - N_W)[1.05 \times 10^{-2} - 2.39 \times 10^{-6} (AE + BE^2) - N_W]^{1/8} \quad (9)$$

where $N_W = W^* CD$

E = exposures in MWD/T for all defined regions for a particular fuel.
W* = corrected power factor in all defined regions.

$$\alpha = f(I\&E) \alpha_i (I\&E) + (1-f) \alpha_i I\&E^2$$

In the event of a discharge during the outage (Disch > 0) the discharge correction is applied to the samarium and promethium as it was in the neptunium equations.

Table 4

Symbol	K Reactors		Old Reactors	
	I&E	I&E ²	I&E	I&E ²
A	0.9963	0.839	0.9725	0.806
B	-2.28×10^{-4}	-1.469×10^{-4}	-2.50×10^{-4}	-1.549×10^{-4}
C	0.0901	0.0725	0.1546	0.1307
D	0.0837	0.0935	0.0873	0.0995

Then, the following relationship is calculated using the proper region exposures at a particular fuel level and corrected power factors:

$$C_T = [C_{T(I\&E)}(f) + C_{T(I\&E)2}(1-f)] \text{ c-mk/degree C} \quad (10)$$

Now, B_{MOD}^2 is approximated by setting the sum of all bucklings in the central zone equal to zero:

$$\delta B_{MAT}^2 + \delta B_{Xe}^2 + \delta B_{Sa}^2 + \delta B_{Ne}^2 + \delta B_{LTG}^2 + \delta B_{MOD}^2 + \delta B_M^2 + \delta B_{Leakage}^2 + \delta B_{Control}^2 + \delta B_{Non-Uran}^2 = 0 \quad (11)$$

⁷⁷ HW-70780, GF Bailey, Ibid

⁷⁸ HW-2158 Personal Notebook, R.A. Chitwood (Pages 67-68-69)

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$\begin{bmatrix} A \\ B \\ C \\ D \end{bmatrix}$ = corresponding parameter values in Table 4.

As in the case of neptunium, the nonequilibrium equation is solved twice if the operating period is less than eight days ($T_o(Z) < 8$).

The promethium equations used by the program are:

K Piles

$$P_m = (1.557 \times 10^{-3} W^*) \mu b$$

Old Piles

$$P_m = (2.468 \times 10^{-3} W^*) \mu b$$

(These two equations give the promethium effect in "microbuck" of potential samarium.)

Other factors in the samarium equation (Eq. (6)) are defined as follows: ^{7/9/}

$$\lambda_p = \text{decay constant of promethium} = 0.3072/\text{day}$$

$$\overline{PF} = \frac{p}{\lambda_p - \sigma_s \phi} = \text{production factor}$$

$$\sigma_s = \text{the samarium thermal absorption cross section } (6.6 \times 10^{-20} \text{ cm}^2)$$

$$\phi = \frac{P \cdot 1.21}{\alpha} \quad (9)$$

The factor 1.21 corrects for the nonuniform front-to-rear flux distribution.

$$P = \frac{W^*}{\text{Number of tubes in the pile}}$$

α is a conversion factor relating tube power and absolute flux

Representative values are given as follows:

K Reactors

α_1

$$I\&E \quad 4.9 \times 10^{-11} \text{ KW}/(\text{Neutron}/\text{cm}^2/\text{sec})$$

$$I\&E^2 \quad 5.5 \times 10^{-11}$$

Old Reactors

$$I\&E \quad 4.3 \times 10^{-11}$$

$$I\&E^2 \quad 4.6 \times 10^{-11}$$

Using this trial solution for δB_{MOP}^2 , a corresponding first approximation for the effective graphite temperature is found as follows:

$$T_1 = \frac{\delta B_{T_1}^2}{C_{T_1} C_{b/k}} \quad \text{degrees C}$$

where $C_{b/k}$ is the conversion factor, microbucks to c-mk. Then, the first approximation for the power-temperature relationship can be found for each region:

$$K = \frac{T_1}{W_c^*} \quad \text{degrees C/MW}$$

where W_c^* is the total power (in MW) generated in the central zone.

Now, the graphite temperature in exterior regions can be found, assuming K to be a generalized power-temperature coefficient:

$$T_i = (KW_i^*) \text{ degrees C}$$

where W_i^* is the total power generated in region i .

Finally, δB_T^2 can be calculated for extension regions:

$$\delta B_{T_i}^2 = \left[\frac{C_{T_i} T_i C_{b/k}}{C} \right] \mu b$$

Now, since all the bucklings are known for every region, the critical equation can be solved to determine the uniform δB^2 required for a critical reactor. Programmed corrections in the regional $\delta B_{T_i}^2$ are made until the uniform buckling requirement converges to zero.

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