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AUTHOR

L. W. Lang
&
W. S. Nechodom

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E-N AND BLANKET CONVERSIONS
FROM ANALYSIS OF TUBES IRRADIATED AT H

by

L. W. Lang

and

W. S. Nechodom

February 10, 1961

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E-N AND BLANKET CONVERSIONS
FROM ANALYSIS OF TUBES IRRADIATED AT H

INTRODUCTION

This document interprets the analytical data obtained on recent E-N test loadings at H Reactor and recommends methods for applying the data to future test loadings.

SUMMARY

A block-loading of striped columns and tubes simulating a blanket loading were analyzed for product after irradiation under Production Test IP-255-A-9-FP⁽¹⁾. An average of the Pu yield from three striped central tube columns was .772 g/MWD at 746 MWD/T. The tritium produced from four such central columns was .235 Pu g atom equivalents/MWD. The yield from an E-metal column adjacent to the blanket was .679 g/MWD at 746 MWD/T. The tritium yield in the blanket was 1.407 g atom equivalents per adjacent tube MWD.

These results have been rationalized to full-pile values. It is concluded that blanket loadings for well-flattened reactors such as H and C Reactor will give increased pile conversion ratios of .049 g atom equivalents/MWD. The previously estimated value was .02 g atom equivalents/MWD. At the older reactors, somewhat reduced values will be obtained. For the E-N loading at H, pile gains of .149 g atom equivalents/MWD were indicated. Previous documentation had calculated these values as .096 g atom equivalents/MWD. These data are shown in Tables No. 3 and 4.

The above yields reflect a reduction of .031 g atom equivalents of tritium per MWD from measured values to compensate for rod and flattening losses not sustained in the block loading.

Although the H-Pile test results are higher than values previously calculated, they have been thoroughly analyzed and are believed to be representative. The conversion ratios obtained by test result are probably better than the calculational methods available for theoretical determination. Such calculations must determine the physics parameters for inhomogeneities within individual tubes for both thermal and epi-thermal reactions. The calculated values for individual tubes must in turn be extrapolated to represent full pile irradiation.

If theoretical calculations are made in a consistent manner, however, they can be normalized against the H-Reactor data to give consistent values for other reactor and fuel systems. Methods for extrapolating the data are given in the Appendix as well as a more detailed critique of individual tube data.

(1) HW-60340, "E-N Demonstration Test," Hall, RE and Nechodom, WS, 5/8/59.

DISCUSSION

General Physics Background

The phenomena of neutron leakage, both fast and thermal, from a reactor is well recognized, but some of the implications are not completely understood. The reactor can be roughly divided into a flattened zone in the center of the pile and a buckled zone in the fringe. These areas would look like a bull's-eye imposed on a front-face temperature map of the reactor. The central zone would correspond to the center circular target and would have relatively constant tube powers. The fringe zone outside of the "target" would have decreasing tube powers and is referred to as the buckled zone, see Sketch No. 1.

SKETCH NO. 1

If, for some reason, it is wished to maintain pile power but decrease the central tube powers, it could be accomplished if we could increase the "target" or central zone. Actually this is accomplished to a greater or lesser extent by radially enriching the reactor and adding .947 enriched U columns in a location indicated by the dotted line. Each pile has different amounts of E-metal columns and a varying number of central zone tubes. Thus C and H Reactor have as many as 1650-1700 effective central tubes (pile power/central tube power) and some of the older reactors have as few as 1400-1500. Since pile efficiency is a function of the number of ruptures that occur, and this in turn is considered an exponential function of the tube power, it has appeared desirable in the past to operate with as many ECT's as possible. But -- there's a hooker!!

Those tubes outside of the bull's eye, lose neutrons by leakage and their conversion ratio will not be as high as those inside. For instance we can consider a tube on the sloped curve of the buckled zone. Its production is a function of all of the tubes around it. Thus, its production depends largely upon the neutrons generated in adjacent columns. But the production of neutrons

General Physics Background (Cont'd)

is a function of the tube power. Thus although the power in the adjacent tube nearest the target is higher, the power in the adjacent tube nearest the pile edge is lower. The average of the two does not equal the power of the tube being considered. Thus, the net conversion is lower in this tube than it is in a central tube. The nearer the edge of the pile, the more pronounced this effect becomes until at the very edge of the pile the conversion ratio may be as much as 10 percent lower than in the center.

The actual reduction in conversion is a function of the curvature of the slope. Thus, for a well-flattened pile, the loss is greater than for one that has a modest number of effective central tubes.

This phenomena can now be looked at in terms of added radial enrichment. Enriched U columns supply more neutrons than they consume. Unfortunately, a good percentage of these are lost to leakage. Thus, an E-metal column charged as indicated in Figure No. 1 might use half of the excess neutrons in bolstering the central tubes, but the other half would be lost to leakage.

If we now considered putting in a neutron absorbing material in the last lattice unit of the pile, we could capture some of those neutrons that were formerly lost.

The greater the flattening that had been achieved prior to our putting in the blanket, the greater the capture of neutrons in the blanket. Unfortunately, the pile would probably not run because the blanket would absorb too many neutrons. Thus, additional enrichment would be required to replace the neutrons absorbed. The reactor with additional enrichment charged might look like Sketch No. 2.

FRONT FACE MAP OF
BLANKETED REACTOR

NOTE: Incremental Power Increase to Sustain Reactor Power (With Blanket).

POWER PROFILE ACROSS AA'

SKETCH NO. 2

General Physics Background (Cont'd)

It should be emphasized that some leakage will probably still occur. Thus, by adding further enrichment, the steepness of the slope of the flux curve and the leakage is increased. Although most of the neutrons would be absorbed in the blanket (depending on how "black" and how big the neutron absorber is), some will still leak from the reactor.

This background is essential to understanding the conversion data obtained, and to making practical application of the data. We wish to derive conversion ratios for the "Blanket Loading", and for the full-pile E-N loading that incorporates the blanket with a striped core of alternated enriched U segments and lithium aluminum targets.

Three sources of conversion ratio information will be used:

- (1) The analytical results from Production Test PT-IP-255-A-FP(2).
- (2) The Accountability Manual(3)
- (3) Physics constants and conversion ratio data calculated by the Flex-1 Program of R. J. Shields and R. O. Gumprecht.

We wish to rationalize the data from the test, accepted values of Reference 2, and the calculations from the Flex-1 program. The data is considered good as reported by Nechodom⁽⁴⁾ both in Pu and tritium from a center "striped" column. Data is also given for the tritium from a blanket column and for the Pu obtained in the immediately adjacent supporting .947 percent U-235 enriched U column. These values and supporting data are given in Table No. 1.

These data can be compared with that obtained from the Accountability Manual and the Flex program as shown in Table No. 2. However, all of this data assumes full-reactor conversions and thus averages the conversion ratios of all individual tubes. They contain both external leakage losses (neutrons that get out of the reactor) and internal leakage losses (neutrons lost to control rods and flattening).

The test loading, on the other hand, was an isolated square of columns, and thus did not lose neutrons to rods and flattening columns. The actual test values can be adjusted, however. Roughly 12 mk of reactivity is contained in the rods and flattening columns. By adjusting only the tritium side of the conversion ratio, the corrected test values would be .772 g Pu/MWD plus .204 g atom equivalents of tritium/MWD. This reduction probably over-corrects

(2) HW-60340, Ibid.

(3) HW-50300-B, "Irradiation Processing Department Plutonium Conversion Tables," Hoba, DW, 9-26-60.

(4) HW-66775, "Semi-Final Report, E-N Conversion Ratios," Nechodom, WS, 12-22-60.

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TABLE NO. 1

CONVERSION AND SUPPORTING REACTOR DATA

Central Zone

Striped E-N Columns
@ 746 MWD/T

.772 g Pu/MWD
.235 Pu g atom equiv.
Tritium/MWD

Fringe Zone

Enriched U columns adjacent to Blanket
@ 746 MWD/T

.679 g Pu/MWD

Blanket

Yield in lithium based on MWD of the
adjacent E-metal column

1.407 Pu g atom equiv.
of Tritium/MWD

Percent Pile Power

E-Metal of Nat U Loading

15.1 in 4th and 5th lattices

Blanket Load*

2nd Lattice Unit
3rd Lattice Unit
4th Lattice Unit
5th Lattice Unit
6th Lattice Unit

4.34 from Enriched U
5.99 from Enriched U
7.63 from Enriched U
7.50 from Enriched U
4.58 from Enriched U
(6th Lattice Unit not completely
charged with Enrichment)

Total ECT in Blanket Loaded Reactor

1660

*Values are extrapolated from the radius used as a test to the full reactor.

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TABLE NO. 2

ACCOUNTABILITY REFERENCE AND FLEX-1 VALUES

<u>Acct. Reference for Old Reactor</u>	<u>Initial Conv. Ratio</u>	<u>746 MWD/T Conv. Ratio</u>	<u>800 MWD/T Conv. Ratio</u>
Enrichment Columns	.794	.711	.707
Natural U Columns	.972	.831	.824

Flex-1 for Old Reactor

Enrichment Columns	.819	.735	.731
Natural U Columns	1.000	.859	.855

Reactivity Data from Flex-1

<u>Reactor</u>	<u>Reactivity with .711 Enrichment</u>	<u>Reactivity with Present Enrichment</u>	<u>% E-Metal Currently Used</u>	<u>Reactivity with Full Enrichment .947 Material</u>
Old	3.7 mk	20	6	102.1 mk
C	.1	30	19	100.8
K	3.6	20	14	96.0

General Physics Background (Cont'd)

for the internal leakage, but will be used in the balance of the calculations. If these values are extrapolated to zero exposure, a reasonable conversion ratio of .204 Pu g atom equivalents of tritium and .855 g atoms of Pu is obtained per MWD. These total to an initial conversion of 1.059 vs. present accepted values of 1.000 or slightly less as the over-all pile ratio. If we now assume a linear relationship over the buckled zone from the .679 to a .772 in the center of the reactor, we can use .726 g Pu/MWD @ 746 MWD/T for fringe charged E-metal. The indicated average leakage from the buckled zone if expressed as lost conversion, would be the difference between the two yields, .772 - .726 or .046 g/MWD.

This loss in conversion may be largely regained by neutron capture in the blanket; but as tritium. This leakage term when added to the tritium conversion expected from added enrichment columns amounts to a value of .0373 g atoms of tritium conversion per reactor MWD. This does not allow, however, for leakage from enrichment columns used for reactivity prior to the blanket loading. From the actual conversions obtained at H, it would appear that in addition to the above conversion, more than fifty percent of the excess neutrons furnished by the radial enrichment at H were lost to leakage but captured in the lithium aluminum when blanket loaded.

From the above considerations, then, a conversion ratio can be calculated for both the blanket and an E-N load. These values are shown in Table No. 3 for H-Reactor. Where reactors are less flattened than H-Reactor, the gains will not be quite as large for the blanket loadings, inasmuch as additional amounts of enriched U will be required to secure the same production. With this in mind, conversions have been given for several reactors, before and after adding blankets. A more detailed method of extrapolating the data from the H-Reactor test to other reactors is given in the Appendix.

Final values are corrected for decay of tritium as well. Allowance of one percent is made for tritium from core loadings and four percent for blanket loads.

Where reactors contain less enrichment than H-Reactor, the conversion ratio gain will be somewhat reduced if the number of ECT is held the same as at H-Reactor (1660). In Table No. 4, approximate values are shown for an old reactor holding six percent in enrichment and C-Reactor holding roughly nineteen percent.

Analytical Extraction Data

The test details and extraction data is detailed in References 5 and 6. The blanket data was secured from columns irradiated at H-Reactor in an upper portion of the pile as shown in Sketch No. 3A. The striped column data (E-N loading) was obtained by product analysis from the hatched tubes within a tube grouping at H Reactor as shown in 3B.

(5) HW-60340, Ibid.

(6) HW-66775, Ibid.

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TABLE NO. 3

CONVERSION RATIOS FOR BLANKET,
E-N AND NATURAL U LOADS AT H REACTOR

	Percent <u>Pile Power</u>	<u>Conversion Ratio</u> @ 746		<u>Contribution</u> <u>to Reactor</u> <u>Conversion Ratio</u>	
		<u>Pu.</u>	<u>Tritium</u>		
A. <u>Natural U</u>					
Central Nat U	69.9	.841*		.588	
Enrichment	15.1	.711		.107	
Nat Fringe	15.0	.801*		<u>.120</u>	
TOTAL					.815
B. <u>Blanket Loaded</u>					
Central Nat U	69.9	.841		.588	
Enrichment	30.1	.726	.192**	<u>.276</u>	
TOTAL					.864
C. <u>E-N Load</u>					
Central Stripe	69.9	.772	.202**	.688	
Blanket	27.8	.726	.192	<u>.276</u>	
TOTAL					.964
Increased Conversion Blanket (g atom Pu or Equiv/MWD)					.049
Increased Conversion E-N (g atom Pu or Equiv/MWD)					.149

*Assumes accountability numbers with adjustment for leakages.

**One percent decay in core, four percent in Blanket.

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TABLE NO. 4

NAT U CONVERSION RATIOS OF OLD PILE
(OTHER THAN H) AND C-PILE

	<u>Percent</u> <u>Pile Power</u>	<u>Conversion Ratio</u> <u>@ 740 MWD/T</u>	<u>Contribution to</u> <u>Pile Conversion</u>
A. <u>Old Reactor</u>			
Central Core	69.9	.841 g/MWD	.607
Enrichment	7.0	.711 g/MWD	.050
Nat Fringe	23.1	.801	<u>.185</u>
TOTAL			.823
B. <u>C Reactor</u>			
Central Core	69.9	.841	.588
Enrichment	19.0	.711	.135
Nat Fringe	11.1	.801	<u>.089</u>
TOTAL			.812
C. Blanket Loading (H-Pile Results)			.864
D. E-N Loading (H-Pile results)			.964

Analytical Extraction Data (Cont'd)

TOP OF UNIT		
L1 Al		L.U. 1
Enr.		L.U. 2
Enr.		L.U. 3
Enr.		L.U. 4
Enr.		L.U. 5
60% Enr.		L.U. 6
Nat		L.U. 7

SKETCH NO. 3a
(BLANKET LOAD)

Nat	EN	EN	EN	EN	Nat	Nat
EN	EN	EN	EN	EN	EN	Nat
EN	EN			EN	EN	Nat
EN	EN			EN	EN	Nat
EN	EN	EN	EN	EN	EN	Nat
Nat	EN	EN	EN	EN	EN	Nat

SKETCH NO. 3b
(STRIPED CORE)

Hatched
Central
Tubes
Used For
Analytical
Data

The tritium produced in the lithium aluminum (both loadings) was measured at Savannah River. For the blanket loading the yields were related to the power of the second lattice unit of slightly enriched U and thus are expressed in the document as g atom equivalents per MWD of the adjacent column (L.U. 2). The power of the tubes in the test are related to full pile by assuming that they are representative of a cylindrical pile radius containing the same number of tubes as an old HAP0 reactor.

The calculated values for the successive lattice units of enrichment are given in Table No. 1.

It was assumed that the Pu formed in the enriched columns was an average of that formed in the lattice adjacent to the lithium aluminum and that measured in the center of the pile. Individual tube powers obtained from venturi and outlet temperatures were verified by checking with production records. In addition, the U-235 burn-out values correspond generally to the stated power generation. Tritium measurements at Savannah River have an automatic check in that a gas-balance can be made between helium and tritium found in the lithium. The values of individual tube extractions has been given in previous documentation by Nechodom⁽⁷⁾.

Comparison With Previous Calculations

The conversion ratio gains claimed in earlier references^(8,9) for blanket and E-N loadings did not take as detailed a look at the phenomena of leakage as has been done in this draft. Also, higher values were assumed at that time for natural U reactor conversions. If the exposures of that document are reduced to a uniform 746 MWD/T, the gains would be concluded as .023 g atom

(7) HW-66775, Ibid.

(8) HW-62462, "Production Economics Associated with a Supported Blanket Load," Lang, LW, 10-15-59.

(9) HW-59959, "HAP0 Operating Parameters for Combined Operation Model," Greager, OH, 4-10-59.

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Comparison with Previous Calculations (Cont'd)

Pu or equivalent/MWD increase in conversion ratio based upon 29 percent additional enrichment being required. From the H-Reactor test (when extrapolated to old reactors), a gain of .041 g atom Pu or equivalent/MWD is concluded for an additional twenty-three percent enrichment. A .052 gain would be indicated at C Reactor.

The predicted production parameters of a full E-N load are given in Reference 5. In that reference the gain from old reactor E-N loading was calculated to be 12.8 percent increase over a natural U load. By extrapolating the results of the H-Reactor test, 17.1 percent is indicated.

L. W. Lang, Principal Engineer
Manufacturing Operation
IRRADIATION PROCESSING DEPARTMENT

W. S. Nechodom, Engineer
Research & Engineering Operation
IRRADIATION PROCESSING DEPARTMENT

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APPENDIX A

CRITIQUE

Single column conversion data would only occasionally fortuitously typify a full reactor loading since any given metal column is strongly influenced by the nature of the surrounding lattices. Plutonium production is a function of both thermal and epithermal flux, but the power of the uranium is primarily a function of thermal flux.

In normal E-metal the relative amount of epithermal to thermal neutron capture to form plutonium is about .56 to 1. The amount of thermal neutron capture to form plutonium in a given lattice is readily calculated by using thermal cross sections to relate captures in U-238 to fission events and therefore heat output. The epithermal flux, however, depends on the characteristics of adjacent lattices which (to a fairly good approximation) produce the epithermal neutrons which are captured in the lattice under study. For qualitative reasoning purposes, assume that the epithermal flux in the lattice under study is proportional to the average heat output of adjacent lattices.

Example 1

Consider the case of an isolated E column in a lattice of natural uranium: The initial conversion ratio (in atoms Pu buildup per atom 235 destroyed) for an infinite lattice of any type column is:

$$C_0 = \frac{1}{1+\alpha} \left[\frac{N_{238} \sigma_{a,238}}{N_{235} \sigma_{f,235}} + \epsilon(1-P) \nu_{235} \right]$$

where $(1+\alpha)$ = ratio of total captures to fissions in U-235.

N = number of atoms

σ_a = absorption cross section

σ_f = fission cross section

ϵ = number of fast neutrons produced from all fissions relative to the number of neutrons produced from thermal fissions only.

P = number of fast neutrons which escape resonance capture in U-238.

ν_{235} = number of neutrons produced per fission.

In this form, $\frac{1}{(1+\alpha)} \frac{N_{238} \sigma_{a,238}}{N_{235} \sigma_{f,235}}$ is the thermal portion of the plutonium

yield, and $\frac{1}{(1+\alpha)} \epsilon(1-P) \nu_{235}$ is the epithermal plutonium yield.

It is not a bad approximation for the present purpose to assume that the epithermal to thermal flux ratio is the same for E-metal and natural uranium fuel elements.

It has been observed that the relative power of an E column in a natural uranium lattice is 1.16.

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Example I (Cont'd)

In an infinite E metal lattice, the thermal yield is .4392 atoms Pu/atom 235 destroyed and the epithermal yield is .2448 atoms Pu/atoms 25 destroyed. If it is assumed that the epithermal flux in an isolated E column is reduced to the natural uranium level, then the E column epithermal conversion ratio becomes $.2448/1.16 = .211$ and the initial conversion ratio of an isolated E column relative to the conversion ratio in an infinite lattice is

$$\frac{.4392 + .211}{.4392 + .2448} = .951, \text{ or } 5\% \text{ lower.}$$

Example II

Consider the case of the E columns located at the edge of the reactor in an E-N blanket load: Qualitatively, it may be seen that this row of E columns "sees" epithermal neutrons from only one side, or about half the normal number of epithermal neutrons. However, the tube powers of the inwardly adjacent tubes are higher than the tube in question, and therefore this "half" is increased by the ratio of the adjacent tube powers to the power of the tube being considered. Let this ratio be 1.47/1.

Now the initial conversion ratio for this column relative to the conversion rates in an infinite lattice can be estimated to be:

$$\frac{.4392 + 1/2 \times 1.47 \times .2448}{.4392 + .2448} = \frac{.6192}{.684} = .905 \text{ or } 10\% \text{ lower than normal.}$$

The effects illustrated in these examples would be accentuated if conversion ratios at goal exposure were compared, rather than initial conversion ratios, since any plutonium burnout is essentially a function of thermal flux only.

Values for E metal conversion ratio taken from the current accountability tables (HW-50300-B) tend to be adjusted to observed values of E metal conversion ratio (very little data is available). These data have come from E metal columns, which are essentially isolated in natural uranium loadings. That portion of the data which was derived from fringe E loadings was at least partially effected by the "edge" effect illustrated in Example II.

It is therefore obviously not fair to directly compare E metal conversion ratios obtained from a block loading such as the E-N test to these tabulated values which are more-or-less adjusted to conform with present E metal loading configurations. As a matter of fact, if these are compared, a large portion of the difference may be explained by referring to Example I.

Two data points were obtained from analysis of the E-N test; a plutonium conversion ratio of .772 gms/MWD for the central zone, and a .679 gms/MWD at the same exposure for an E column adjacent to the lithium aluminum blanket. ($.679/.772 = 0.88$). Note the similarity of this ratio to that obtained in Example II. It is thus reasonable to assume that the measured plutonium conversion ratios obtained in the E-N test are good values, and are indeed better direct experimental data than exists for the bases for any other conversion values presently in use.

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CRITIQUE (Cont'd)

A significant problem which remains is establishing conversion ratio values for current reactor loadings which will be equivalent to those obtained for the E-N load by experiment. Again, values contained in current accountability documents have been more-or-less normalized to observed data; however, in the case of natural uranium, the normalization process does not appear to have caused nearly as large a deviation from strictly theoretical values. It therefore appears reasonable to use these tabulated values as is, provided that no further downward adjustment is made to compensate for the "edge effect" illustrated in Example II; it further appears reasonable to use current E-metal values from the tables, recognizing that the over-all pile conversion ratio so obtained should be, if anything, a little higher than actual observed CPD results, thus tending to yield conservative values for the indicated incentives for E-N loadings.

EXTRAPOLATING E-N RESULTS TO CHANGED REACTOR AND METAL LOADINGS

If we assume that $.204$ Pu g atom equivalents of tritium is the correct conversion ratio for H-Reactor core loading and that a length ratio of one-inch of lithium per 15 inches of $.947$ enriched U is required for striped columns, the production parameters of E-N and blanket loadings in other reactors can be estimated.

The following qualitative relationships appear evident:

- (1) The amount of tritium made by adding enrichment to a reactor will be proportional to the reactivity difference between the reactor in its normal state and a reactor fully loaded by E-metal. i.e., A difference of 98.4 mk is calculated by the Flex-1 program for natural U loadings and slightly enriched U loadings at H-pile. With this reactivity difference and appropriate length of E-metal to lithium aluminum, $.204$ g atom equivalents of tritium can be made.
- (2) The thermal utilization of the lithium will be improved over that of enriched U and thus the thermal utilization of the reactor will be improved.
- (3) The resonance escape probability will be increased by removing some of the U-238 of the reactor.

A first approximation must be made of the required length ratio of lithium-aluminum to enriched U segment. To make this calculation the difference in reactivity between a natural U load and $.947$ enriched U load can be calculated by the Flex program. The ratio of lithium to column length should be proportional to the reactivity difference. Thus for H-Reactor, a length ratio of 1 to 15 appeared to give balanced reactivity. The reactivity difference between natural and E-metal was 98.4 mk. E-N core conversion can be related to the H-Reactor test results by the following calculational procedures.

1. Estimate a value for the length ratio of E-metal in an E-N striped load:

$$R = \frac{1.476}{1.476 + k_E - k_{base}}$$

where k_{E} is the k_{eff} for a full-pile loading of E metal and k_{base} is the k_{eff} of a standard central zone loading which provides adequate operating reactivity.

2. To obtain pen:

$$P'_{EN} = [1 - (1-p')_{EN}]$$

$$(1-p)_{EW} = (1 - p')_E \times R$$

3. To obtain f_{EN} :

$$f_{EN} = \left[1 - (1 - f)_{EN} \right]$$

$$(1 - f)_{EN} = \frac{1 - f}{R}_E$$

- $$4. \quad k_{\infty EN} = N_E \epsilon_{E \dot{p}' EN} f_{EN}$$

5. Atoms T produced:

$$N = 100 (k_{\infty EN} - k_{\infty base}) = \text{atoms T/100 thermal neutrons absorbed in fuel.}$$

6. Initial conversion ratio in E-portion of the load:

$$Co = \frac{1}{(1 + \alpha)} \left[\frac{N_{238}}{N_{275}} \frac{\widetilde{a}_{238}}{r_{235}} \right] + \frac{1}{(1 + \alpha)} \left[\epsilon_E (1 - PEN) v_{235} e^{-b^2 L^2} \right]$$

where the first term (X) is the build-up in the thermal neutron energy region; atoms of plutonium per atom 235 destroyed.

The second term (y) is the atoms plutonium built up in the epithermal neutron energy region per atom of ²³⁵ destroyed in the thermal term.

$$(1 + \alpha) = \frac{a_{235}}{f_{235}} \approx 1.2062$$

$$\gamma = 2.47 \text{ for } U^{235}$$

$e^{-b^2_L \tau}$ = The fast neutron non-leakage probability in the longitudinal dimension only.

Calculate the Co; let the thermal term = x
 " " epithermal term = y

7. In the thermal term, the plutonium atoms formed per 100 thermal neutrons absorbed in fuel

$$= \left(\frac{X}{1 + X} \right) 100 = Z$$

The U²³⁵ atoms destroyed = 100 - Z.

8. In the epithermal term, the number of plutonium atoms built up = Y (100 - Z)
9. Convert Co to grams Pu/MWD:

$$\text{gms/MWD} = \frac{\text{Co} \times 1.05418}{N_1} = \text{ICR}$$

where N₁ is obtained from the Flex-1 program which is used to obtain the other parameters.

10. Calculate the tritium build-up in grams Pu equivalent:

$$T = \frac{N}{Z+Y} \frac{\text{ICR}}{(100-Z)} = \text{gms Pu equiv./MWD}$$

11. Normalize to H-Reactor test data:

$$T_{\text{normalized}} = 1.162 T_{\text{calculated}}$$

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BLANKET LOADING CONVERSION RATIOS

Extrapolation of H-Reactor blanket results to blanket loadings at changed reactors and metal loadings require buckling calculations. The general procedures for making such calculations and normalizing the theoretical values to H Reactor data, is given below:

- A. Assume that the extrapolated pile boundary is at the outer edge of the Li-Al lattice.
- B. Assume a flat zone radius equal to that of the natural U loading. Using cylindrical geometry, determine the radial geometric buckling of the fringe region. By trial and error, adjust the flat zone radius until the radial geometric buckling + 15.2 equals the material buckling of the E-metal.
- C. The total number of neutrons leaking radially out of the reactor:

$$N_1 = \frac{k_{E} - k_{E}}{1 + b_g^2} \times 100 = \text{neutrons/100 thermal neutrons absorbed in fuel.}$$

where b_g^2 = the radial geometrical buckling found in B.

- D. Since the k_{eff} of all parts of the reactor must equal 1.00, then an additional number of neutrons must be captured in the Li-Al:

$$N_2 = \left[\frac{k}{1 + b_g^2 \text{ total}} - 1 \right] \times 100$$

where $b_g^2 \text{ total}$ = the radial geometrical buckling + 15.2 b
(longitudinal buckling)

- E. Assume that $N_1 + N_2 = N_T$ = the total number of neutrons which could theoretically be captured in the Li-Al.

$$F. Co = \frac{1}{1 + \epsilon} \left[\frac{N_{28} \sigma_{a28}}{N_{25} \sigma_{i25}} \right] + \frac{1}{1 + \epsilon} \left[\epsilon (1 - p) \sqrt[25]{e^{-b^2 g \tau}} \right]$$

where $e^{-b^2 g \text{ total } \tau}$ = non-leakage probability of fast neutrons in all dimensions.

- G. The number of plutonium atoms built up in the thermal term:

$$\frac{X}{1 + X} 100 = Z$$

The number of plutonium atoms built up in the epithermal term:

$$y (100 - Z)$$

The initial conversion ratio in the E-metal is:

$$\frac{Co \times 1.05418}{N_1} = \text{ICR, gms Pu/MWD}$$

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H. Calculate the theoretical tritium build-up rate:

$$T = \frac{(N_1 + N_2) \times \text{ICR}}{Z + y(100-Z)} = \text{grams Pu equiv./MWD in the E-metal blanket}$$

I. Normalize to H-Reactor test data:

$$T_{\text{normalized}} = .97 T_{\text{calculated}}$$

The theoretical number of neutrons leaking into the blanket (and from the reactor) by the above calculation when compared to the H-Reactor measured tritium formation is 1.0 to .97. Thus to normalize theoretical calculations to H-Reactor results, multiply by .97.

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8/29/94

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