Technical Information Series

BNWL-152
45-

AEC
RESEARCH
and
DEVELOPMENT
REPORT

PROGRESS REPORT
PHOENIX FUEL PROGRAM
FY-1965

P. L. Hofmann

Due for		60008-703	FEB 1 1966
C. H. Hochman		50008-703	August 1965
		R403	
		File	
		Billy

BATTELLE-NORTHWEST
RICHLAND, WASHINGTON

PACIFIC NORTHWEST LABORATORY operated by BATTELLE MEMORIAL INSTITUTE

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

PACIFIC NORTHWEST LABORATORY

RICHLAND, WASHINGTON

operated by

BATTELLE MEMORIAL INSTITUTE

for the

UNITED STATES ATOMIC ENERGY COMMISSION UNDER CONTRACT AT(45-1)-1830

PRINTED BY/ FOR THE U. S. ATOMIC ENERGY COMMISSION

3 3679 00060 1726

BNWL-152

UC-80 - Reactor Technology

TID-4500

PROGRESS REPORT - PHOENIX FUEL PROGRAM - FY-1965

Compiled By:

P. L. Hofmann

Principal Contributors:

D. G. Albertson
J. B. Burnham, Jr.
G. J. Busselman
J. C. Fox
C. M. Heeb
R. H. Holeman
P. L. Hofmann
J. J. Regimbal
K. G. Toyoda

FIRST UNRESTRICTED
DISTRIBUTION MADE

NOV 12 '65

August 15, 1965

PACIFIC NORTHWEST LABORATORY

RICHLAND, WASHINGTON

FIGURE 1

Reactivity - Lifetime Characteristics: Use of B-10 as Burnable Poison

FIGURE 2

Reactivity - Lifetime Characteristics: Effect of Initial Fuel Loading

I. Pu Value in Hx-Pu Burners

An important factor in the evaluation of Pu-burners for portable power packages or in power units for naval or merchant vessels is their relative fuel cost compared to fully enriched U-235 reactors. One way of making this comparison is to obtain the Pu indifference value based on an assumed U-235 price.

In BNWL-CC-119⁽³⁾, Pu values for a range of parameters have been estimated. Some of the assumptions which were made are rather speculative, and hence the results obtained must be considered quite preliminary in nature. The main purpose of this study was to serve as the starting point of a more detailed economic evaluation of the potential of Pu burners for special purpose applications.

An idealized PWR served as the basis for this evaluation. The reactor uses zircaloy as the structural material and fuel diluent. The core volume is a nominal 500 liters with an effective metal-to-water ratio of unity. The fuel elements are pictured to be plate-type. The fuel meat could be a fuel-zircaloy alloy or a fuel oxide dispersion in zircaloy.

The Hx-Pu fuel is assumed to have an isotopic composition consisting of 50 a/o Pu²³⁹, 31 a/o Pu²⁴⁰, 17 a/o Pu²⁴¹ and 2 a/o Pu²⁴². Nuclear calculations for the Pu reactors have been taken from HW-79977⁽²⁾. Calculations for comparable, fully enriched (93 a/o)U²³⁵ cores were also performed.

The Pu indifference value was determined by setting the fuel cycle costs of the Pu reactor equal to those of the U-235 reactor. The simplifying assumption was made that reprocessing costs, fuel shipping costs, etc. are equal for the Pu and U-235 cases. Also no credit was taken for the production of trans-plutonium isotopes in the Pu cores. Consequently, only

three cost items were considered:

1. Fuel fabrication costs and associated interest charges.
2. Fuel inventory charges.
3. Fuel burnup charges.

In all calculations it was assumed, that the core life is one year, at a load factor of 8. Fuel use charge, $i = 4.75\%$, and working capital charge, $i = 12\%$ were used. The price of highly enriched U-235 was taken at \$12/gm. Fissile Pu isotopes (Pu-239 and Pu-241) were assumed equivalued. Zero value was assigned to Pu-240 and Pu-242. The low charge of $i = .0475$ might be justifiable if subsidies for the U. S. Merchant Marine are assumed.

The results of the value calculations are shown in figures 3 to 5. For U-235 zircaloy elements, fabrication costs ranging from 200 to 700 \$/ft² of heat transfer area were postulated. A curve for \$100/ft² has also been included for reference. This might be a more representative value for a steel fuel element though the neutronics of the steel core would differ from those of a Zr core. Pu values were obtained assuming 5, 10 and 15% excess charges associated with the fabrication of the Pu elements. For the case of identical Pu and U-235 fabrication costs, the Pu values are unaffected by the unit fabrication cost. For this case the results are tabulated below.

Pu Values; Zero Premium on Pu Elements

<u>Reactor Power</u>	<u>Pu Value (\$/gm fissile)</u>
50	15.8
100	14.9
150	14.5
200	14.3

Examination of Figures 3-5 show that Pu-value is a rather weak function of reactor power, but a relatively strong function of fabrication costs. As

one would expect, the Pu value drops rapidly for high fabrication costs and excessive premiums on the Pu-element fabrication.

It would appear that Hx-Pu elements containing large amounts of fertile material (e.g., Pu-240) have substantial economic potential compared to fully-enriched U-235 elements in Zr/H₂O reactors.

FIGURE 3

Pu-Value: 5% Premium on Pu - Elements

FIGURE 4

Pu-Value: 10% Premium on Pu - Elements

FIGURE 5

Pu-Value: 15% Premium on Pu - Elements

II. Analysis of Pu-240 Resonance Absorption Experiments

The calculation of neutron absorption rates in the 1 e.v. Pu-240 resonance is of prime importance in the analysis of Phoenix cores. Experimental values of the Pu²⁴⁰ resonance integral as determined by PCTR reactivity increment measurements are available in the literature.⁽⁴⁾ These results can be used to evaluate the applicability of various methods of calculating the Pu²⁴⁰ resonance integral. Specifically, these values can be used to test the numerical methods contained in the codes ZUT and TUZ⁽⁵⁾. These codes are based on a suggestion by L. W. Nordheim⁽⁶⁾ that the general integral slowing down equations in collision density, $F(u) = \Sigma_t(u)\phi(u)$ lend themselves to direct numerical solution for $F(u)$. The resonance integrals can then be calculated from:

$$I = E_1 \int_0^{\infty} \sigma_{a_0}(u) \frac{F(u)}{\sigma_{t_0}(u) + \sigma_m} e^{-u} du,$$

where E_1 is the energy corresponding to the base lethargy and σ_m is the potential scattering cross section of an admixed scatterer per absorber nucleus. The codes in the present form, contain features which were developed with the U²³⁸ and Th²³² resonances in view and so a test of their applicability to the giant 1 eV Pu²⁴⁰ resonance is of interest. Comparisons of the integral method and the usual NR1A approximation are also of interest since this last approximation is most often used in calculating the absorption at 1 eV in Pu²⁴⁰.

To determine the value of the resonance integral for Pu²⁴⁰, the reactivity difference between pairs of 0.65 inch diameter Pu-Al alloy rods differing essentially only in the Pu²⁴⁰ concentration were measured in the PCTR.

An equation of the form

$$\Delta\rho_i = C \left[(N_B^{240} I_B^{240} V_B - N_A^{240} I_A^{240} V_A) \right]$$

was assumed.

$\Delta\rho_i$ = the observed reactivity difference between rods B and A
of the i^{th} pair. (ϕ)

I^{240} = the effective resonance integral for the rod (barns)

V = the rod volume (cm^3)

To minimize reactivity difference due to changes in thermal absorption in the rods, a cadmium cover was provided for each rod. The epi-cadmium correction is estimated to be 114 barns. This value, plus the sum of the individual calculated values for each Pu^{240} resonance can be substituted in the above equation along with the observed reactivity difference for the i^{th} pair. The experiment thus affords a way of evaluating the methods used to calculate the integrals by applying goodness of fit tests to the data.

Using the numerical solution of the slowing down equation available from ZUT for each rod pair, the resonance integrals are shown in the table below.

Pu-240 Resonance Integral Values

Rod #	N^{240} n/barn-cm	I at 1 eV barns	ΣI above 1 eV barns	Unresolved* barns	Epi Cd 1/v barns	Total barns
1	5.0×10^{-7}	8,259	113	32	114	8,518
2	1.42×10^{-7}	8,210	113	32	114	8,469
3	4.08×10^{-7}	8,074	113	32	114	8,332
4	1.17×10^{-6}	7,712	113	32	114	7,970
5	3.35×10^{-6}	6,865	112	31	114	7,122
6	9.59×10^{-6}	5,312	110	31	114	5,567
7	2.74×10^{-5}	3,453	106	31	114	3,704
8	7.80×10^{-5}	2,045	95	29	114	2,283
8B	2.20×10^{-4}	1,185	76	26	114	1,401

* The TUZ code has been used to estimate the unresolved resonance contribution.

Using the values from the above table, a least squares fit gives the value $C = 0.142 \pm 0.0016$ for the constant in the experimental equation. Figure 6 is a plot of this line along with the 95% confidence level bands on the abscissa for a given single observation of the PCTR reactivity increment. The line fit demonstrates a high degree of consistency between the observed experimental results and the method of calculation.

The confidence level band serves to estimate the experimental uncertainty. If $X = \frac{I_B N_B V_B}{I_A N_A V_A}$, then the upper and lower values on the resonance integral are given by

$$I_t = \frac{1}{N_B V_B} (X_{\pm} + N_A V_A I_A)$$

where $X_{\pm}$ corresponds to the upper and lower band values on X for a given Y.

For the purpose of comparing the "calculated" resonance integral values with "experimental" values, the best experimental value is

$$I_{\text{experimental}} = \frac{1}{N_B V_B} (X + \frac{N_A V_A I_A}{N_B V_B}),$$

where X is the line value corresponding to observed reactivity for that particular absorber nuclei concentration. The integral value I_A for the other rod in the pair is obtained from the calculated value given from the ZUT code. When all the experimental points are used to determine the exact linear relationship, then the line values become the "true values" in the sense that a large number of repeat measurements would yield the value given by the line if the line obtained represents a good approximation to the true model. The "calculated values" are those obtained through the direct numerical solution for $F(u)$ with subsequent evaluation of the resonance integral.

In the next table, a comparison of calculated and experimental resonance integral values as given by Nichols is shown. The wide experimental uncertainty

at lower absorber concentrations explains the discrepancy between the integral values given here and the values quoted by Nichols. The method used in arriving at the quoted values was to obtain a value of C at the low concentrations (rod 1, 2, and 3) and then to use this value in a pair wise comparison with increasing absorber concentration from rod #3. From the uncertainty at low concentration it is apparent that there is a high probability of obtaining an incorrect evaluation of all subsequent absorber integrals. It is of interest to note that the "calculated" values shown in the table are everywhere within the quoted uncertainty of the "experimental" values even though that uncertainty decreases by almost two orders of magnitude with increasing absorber concentration.

COMPARISON OF EXPERIMENTAL AND CALCULATED RESONANCE INTEGRAL VALUES				
Rod	$n/\text{barn-cm}$	$I_{\text{experimental}}$ barns	$I_{\text{calculated}}$ barns	I_{Nichols} Barns
4	3.35×10^{-6}	$7,647 \pm 1,286$	7,122	$6,730 \pm 430$
5	9.95×10^{-6}	$5,826 \pm 456$	5,567	$5,080 \pm 450$
6	2.74×10^{-5}	$3,724 \pm 163$	3,704	$3,300 \pm 370$
7	7.80×10^{-5}	$2,314 \pm 61$	2,283	$1,990 \pm 300$
8	2.20×10^{-4}	$1,389 \pm 26$	1,401	$1,180 \pm 270$

$I_{\text{experimental}} \sim$ the value corresponding to $x = (N_B V_B I_B - N_A V_A I_A)$
 where $x = y/c$ and $N_A V_A I_A$ contains the calculated
 value of the resonance integral.

$I_{\text{calculated}} \sim$ the value calculated with ZUT and TUZ codes

$I_{\text{Nichols}} \sim$ the original resonance integral values as quoted
 by Nichols.

Comparison of the integral method of solution and the NRIA approximation used in the 1 eV resonance yields virtually identical results at the Pu^{240} density range of the experiments.

FIGURE 6

All Points Line
 Reactivity vs Absorption Difference, 95% CL Band Indicated

The numerical procedures embodied in the code ZUT and NRIA approximation when applied to Pu^{240} resonance absorption yield results which are consistent with the Nichols' experimental results. The values of the Pu^{240} resonance integrals at high Pu^{240} concentration quoted in the original experiment are probably too low for reasons which are clear upon subsequent re-analysis made possible by the availability of more powerful means of computing resonance absorption. It is therefore concluded that both the ZUT code, its progeny NIT and GANIT and the NRIA approximation can be used to calculate Pu^{240} resonance absorption until new experimental evidence to the contrary is produced.

Further details of this analysis are given in BNWL-97⁽⁷⁾.

III. PCTR PHOENIX FUEL EXPERIMENTS

An important goal of the Phoenix fuel program is to develop general methods for calculating the nuclear characteristics of plutonium fueled thermal systems. For this purpose a series of plutonium polyethylene experiments has been carried out in the PCTR to supply a basis for checking the methods in use. 5.1 kgs of Pu have been fabricated into 7940 20 w/o Pu in Al discs, 20 mils thick and 1.96 inches in diameter. The three Pu²⁴⁰ isotopic contents were 8, 19, and 27 nominal percent.

As a preparatory step, a fully enriched U²³⁵ experiment was performed, using 40 mil UAl (33 w/o U, enriched to 93.5 w/o U²³⁵) discs sandwiched between polyethylene wafers.

These experiments and their analysis are summarized in the following sections. The possibility of carrying out PCTR experiments on PuAl - Be systems has also been under consideration. The results of these investigations are summarized in Section III C.

A. UAl - Polyethylene Systems

U²³⁵ rather than Pu was used to try out the experimental techniques and calculational methods with a system which has neither the low-lying resonance structure of Pu nor the contamination problems of a plutonium experiment.

The fuel consisted of 40 mil thick by 1.96 inch diameter UAl discs with 33 w/o U, enriched to 93.0 w/o U²³⁵. Between each fuel disc there were fifteen 20 mil polyethylene moderator discs. Some of the poly discs contained 0.8 w/o B. Every disc in the cell had a 0.20 inch diameter hole in the center. The just critical* condition required 0.126 atoms of B¹⁰ per atom of U²³⁵. The H/U²³⁵ ratio for this system was 217.4. The atom densities in the

* Throughout section III "Critical" is used to denote the condition $k_{\infty} = 1.000$.

experimentally critical system ⁽⁸⁾ are given below.

JUST CRITICAL EXPERIMENTAL
COMPOSITIONS
(atoms/barn cm)

	<u>Cell</u> <u>Composition</u>	<u>Fuel Disc</u> <u>Composition</u>	<u>Moderator Region</u> <u>Composition*</u>
H	$5.00 \times 10^{-2} \pm 0.04$		7.92×10^{-2}
C	$2.50 \times 10^{-2} \pm 0.02$		3.96×10^{-2}
Al	$4.62 \times 10^{-3} \pm 0.02$	5.48×10^{-2}	
B ¹⁰	$2.90 \times 10^{-5} \pm 0.03$		4.60×10^{-5}
O	$2.20 \times 10^{-4} \pm 0.02$		3.49×10^{-4}
U ²³⁴	$2.53 \times 10^{-6} \pm 0.04$	3.10×10^{-5}	
U ²³⁵	$2.30 \times 10^{-4} \pm 0.02$	2.73×10^{-3}	
U ²³⁸	$1.45 \times 10^{-5} \pm 0.01$	1.72×10^{-4}	

NOTE: There is associated with each unit cell some 6.24×10^{-3} atom/barn cm of Al can wall. This material is included in the cross section calculations but is not part of the system which has a multiplication of unity.

* Contains both borated and unborated poly discs.

The numbers shown supersede those given in Table III of HW-84369.

The experiment consists of placing a 19 tube hexagon of the experimental lattice in the test cavity of the PCIR and determining by a null reactivity measurement the concentration of boron required for criticality. This method compares the reactivity worth of a section of the lattice to that of a void. Criticality is indicated when there is no difference between these worths. The experimental lattice is some 12 inches in length, but only the center six inches of the center tube is removed for the null experiment. The remaining

discs act as spectrum buffers and assure a flat flux across the test volume. Fairly extensive analysis of the experiment were carried out and are described in BNWL-CC-148⁽⁹⁾.

The calculational methods used progressed from the ZODIAC code diffusion methods employed for survey type calculation, to a more exacting multigroup transport code. The thermal absorption and fission cross sections in all cases were calculated using a 20°C polyethylene kernel and the epithermal cross sections, i.e., above 0.683 eV, were generated by means of the GAM slowing down code. Thermal scattering cross sections were obtained using a combination of Wigner and Wigner - Wilkins approximations.

The results are summarized in the following Table.

RESULTS OF CRITICALITY CALCULATIONS

	<u>k_∞</u>	
	<u>Homogeneous</u>	<u>Heterogeneous</u>
ZODIAC, Diffusion Theory	1.0888	1.0587
ZODIAC, Homogeneous with Thermal Group Self Shielding Factors from DP ₃		1.0108
Sn(double S ₃), Transport Theory, 6 Angles	1.0889	1.0359
Sn(double S ₅)		1.0224
Sn(double S ₇)		1.0173
Sn(double S ₁₃)		1.0120
Sn(double S ₁₆)		1.0110
Sn(double S ₃₁)		1.0093
Sn(double S ₁₆) Selected Angles		1.0130
Sn(double S ₃₄) Selected Angles		1.0121

The first two entries indicate that the diffusion theory thermal flux depression in the fuel region is insufficient. The diffusion depression factor is 0.9739 where

the double P_3 method gives 0.8963 using the same cross sections.

The second heterogeneous entry in the table is derived from a homogeneous calculation (i.e., no spatial flux variation) with the thermal cross sections reduced for those materials belonging in the fuel plate. No adjustments were made to the other energy groups.

The transport theory analysis made use of an Sn code and the same set of cross sections as used in the diffusion analysis. The number of double angles was varied from 3 to 31. Two attempts were made at selecting the angular areas of most importance in order to achieve a high degree of accuracy with a lower number of angles. The results are shown in the table and indicate that selection of the "correct" angular set is not straightforward.

The "best" calculated number is 1.0093 employing a 31 double angle, 4 energy group, Sn method. The uncertainty in the experimental result due to uncertainties in the composition is about ± 0.006 . In addition, the uncertainty due to experimental technique is of the order of ± 0.001 . On this basis it would appear that the calculations and the experimental results are in good agreement.

It would also appear that for this particular system the use of diffusion theory coupled with an improved spatial thermal flux calculation gives adequate results ($k_{\infty} = 1.0108$).

B. PuAl - Polyethylene Systems

The PuAl - Polyethylene experiments in the PCTR were similar to the UAl experiment described in III-A. The boron concentration required to reduce the infinite multiplication factor, k_{∞} , to unity was measured for various Pu^{240} concentrations, fuel plate thicknesses, and moderator-to-fuel ratios. The measured values of the boron concentration were compared to theoretical predictions to check the accuracy of calculational methods and parameters. The

principal components for these experiments were Pu-Al (20 wt% Pu), pure polyethylene and borated polyethylene (0.90 wt% natural boron). All three materials were in the form of discs, 1.960 in. in diameter and 0.020 in. thick.

The discs were again stacked into aluminum cans (12.0 in. inside length, 2.0 in. ID, and 1/16 in. wall thickness). The fuel, moderator, and poison discs were stacked in layers in the cans so that continuous slabs of each material were formed. For each system the boron concentration was varied until $k_{\infty} = 1.00$. The table below gives the plutonium compositions. The results of the experiments and calculations are summarized in Figure 7.

<u>PLUTONIUM COMPOSITIONS</u>				
<u>Type</u>	<u>at. % Pu²³⁹</u>	<u>at. % Pu²⁴⁰</u>	<u>at. % Pu²⁴¹</u>	<u>at. % Pu²⁴²</u>
I	90.48	8.19	1.25	0.08
II	77.80	18.89	3.01	0.30
III	67.42	26.86	4.76	0.96

The analyses of these experiments is not yet complete. The calculations summarized here were made primarily to provide pre-experiment information. The exact compositions were not known prior to the experiment; so some discrepancy in reactivity was expected.

The first experiment of the series was the 20 mil, H/Pu of 200, 27% Pu²⁴⁰ case. The calculated B¹⁰/Pu ratio of this system was normalized to the experimental value and the set of normalized calculations was then used as a basis for loading the buffer zones of the other experiments. The discrepancy between measured and calculated values of the B¹⁰/Pu ratio -- which range from 10 to 20 percent, corresponding to about 2.5 to 5 percent in k_{∞} -- has not yet been fully resolved.

The calculations again made use of the polyethylene kernel to obtain thermal cross sections and of the GAM slowing down code to generate epithermal

group constants. These cross sections were then used in a Sn transport code to carry out a four energy group, thirteen double angle calculation.

Various methods were compared for obtaining the Pu^{240} absorption cross section near the 1 eV resonance. For most calculations the NR1A approximation was adopted.

As shown in Figure 7, the discrepancy between calculated and measured B^{10}/Pu is greater for the higher H/Pu systems. The analysis of these very thermal systems should benefit from a more exact representation of the thermalization process. For this purpose the space dependent thermalization code THERMOS is being used. Preliminary results indicate that the discrepancy in k_{∞} can be reduced to less than one percent.

C. Proposed PuAl - Be Experiments

Intermediate spectrum, beryllium moderated reactors are also of some interest in connection with the Phoenix program. Unfortunately no experimental plutonium - beryllium criticality data appears to be available to lend support to exploratory calculations in this general area. PCTR experiments similar to the Pu-Al, polyethylene experiments currently underway would have the following advantages:

1. Less fuel material would be required than for bare or partly reflected full critical experiments;
2. The existing Pu-Al alloy wafers employed in Pu-polyethylene experiments could be used;
3. Beryllium metal discs and triffutes are available at PNL
4. Experience gained in doing the polyethylene moderated experiments would be directly applicable for any extension to beryllium systems.

Survey calculations covering nominal moderator to fuel ratios of from 8000 to 100, and involving plutonium in aluminum alloy disc form have been completed. The plutonium isotopic composition is the same as that used in

FIGURE 7

Characteristics of Critical PuAl Borated Polyethylene Systems

the polyethylene moderated experiments. Instead of using the borated polyethylene discs which would have softened the neutron spectrum to a degree unacceptable in an intermediate system, unit multiplication factors can be achieved by the addition of boron in the form of 20 mil boron-aluminum alloy discs. Figure 8 gives the unpoisoned infinite multiplication constant as a function of the beryllium to plutonium atomic ratio for three isotopic compositions.

The calculated plutonium requirements for various moderator-to-fuel ratios range from $\sim 1/2$ kg for Lx Pu at 8000 Be/Pu atomic ratio to ~ 30 kg of more exposed Pu at very low Be/Pu ratios. Beryllium supplies currently on hand at PNL are adequate at all moderator to fuel ratios, although some machining would be required for some configurations.

It is concluded that PCTR experiments with Pu-Be systems are entirely feasible for moderator to fuel ratios of 1000 or greater, and that lower ratio measurements could be made but would require fabrication of additional Pu.

FIGURE 8

Plutonium Beryllium Systems: k_{∞} vs N_{Be}/N_{Pu}

IV. Evaluation of MTR for a Possible Phoenix Fuel Power Experiment

A Pu burner presents enough new features and problems, (in the areas of physics, materials, and engineering design) that only a full scale demonstration experiment can be considered to provide adequate answers to all of the implied questions. While an existing prototype power reactor facility could perhaps serve as a test bed for a demonstration experiment, the cost of such an experiment is high, and the probability of the initial core design achieving the desired performance is not high in view of the uncertainties involved. It may, therefore, be prudent to first perform an intermediate, lower cost experiment which would reduce the uncertainties, particularly in the nuclear design area, prior to a commitment of a full scale power reactor demonstration.

The MTR has some attractive characteristics for such an intermediate facility:

1. The MTR has already operated with a Pu core in 1958 (Cycle 108).

The Pu inventory at that time was low (~ 3 kg) and the Pu²⁴⁰ content was also low ($\sim 4\%$). While the results of this experiment do not provide adequate information about the behavior of Hx-Pu cores, important operational precedents are available. Techniques for MTR plutonium fuel fabrication are also available and development will be minimized.

Detailed calculations to duplicate the cycle 108 physics have been carried out. The results of this work to date are given in HW-83431⁽¹⁰⁾ and are summarized in section IV A.

2. The MTR has relatively high power capability (~ 40 MW). Large fuel burnups can thus be obtained in a relatively short time.
3. The MTR may be available for alternate experimental purposes in the near future.

The main disadvantage of the MTR as a test bed for potential power cores is its low operating temperature ($\sim 110^{\circ}\text{F}$). For this reason, the MTR Hx-Pu run should be primarily considered to be a burnup experiment, rather than a full scale demonstration experiment. The principal objectives of such an experiment would be as follows:

1. It would demonstrate the neutronics of Hx-Pu as a fuel in thermal-spectrum, water-moderated power reactors, and
2. It would reduce the physics uncertainties in the design of future high-temperature Hx-Pu power cores.

If the experiment is sufficiently well instrumented and properly documented, time dependent reactivity data, power distributions and, possibly, isotopic compositions of select fuel pieces could be made available and compared with predictions from nuclear design calculations.

An evaluation of the MTR for a possible Phoenix fuel power experiment was made and the results were supplied to the AEC in FY 1965. In section IV B a summary of this evaluation is presented.

A. Analysis of MTR Cycle 108 - Low Exposure Plutonium Cores

A full, 27 element plutonium core was burned in the MTR in August of 1958 (Cycle 108). However, the MTR plutonium-aluminum plates contained only 4.3 isotopic percent Pu^{240} , not enough for a significant control augmentation. Nevertheless, this experiment is of interest since it furnishes the opportunity of testing computational methods which may be useful in the design of other plutonium cores.

In HW-83431⁽¹⁰⁾ calculations of the statics and burnup of the two fuel loadings which comprised the MTR Cycle 108 experiment were carried out. A summary of the results is given in the ANS transactions⁽¹¹⁾. The first, a "reactor physics" configuration, was brought to critical with 3.69 kg of plutonium and tested at nearly zero power. The second, a "power core," contained

about 0.25 kg less and was burned for 262 Mwd at powers of 5, 20, 25, and 30 Mw for 14, 138, 76, and 34 Mwd, respectively.

Physics statics tests were conducted on several small critical cores in the process of obtaining a full 3 x 9 (27 element) core. The first full core, the reactor physics core, was tested likewise and not burned. The elements used in this core (excepting shim rod follower fuel) each contained the same plutonium load, with only slight variations. Many of these elements were incorporated in the second full core, the power core, in addition to some elements of lower fuel content which were placed so that some power flattening would result. The schedule of high power operation of this core took into account some heat transfer limitations due to high density inclusions which were found in the fuel plates, and the choice of a 14 wt % alloy which is near the eutectic point.

One modification which was required in order that the reactor might accommodate the plutonium loadings and be operated safely was a reduction in the shim rod drive speed. This is necessary because of the smaller fraction of delayed neutrons produced in plutonium fission than in uranium fission.

Several problems arose in the formulation of a reactor model for carrying out calculations. The conditions existing in the reflector were difficult to specify due to several factors. First, years of operation had changed the beryllium properties, and it was felt that the buildup of lithium-6 would about offset the positive effects of n-2n reactions. Secondly, records concerning the samples contained in the reflector pieces are all but unobtainable. Furthermore, some changes were made in the reflector, especially in the lattice positions neighboring the core, after the reactor physics core was tested. Some difficulty was also experienced in providing a model of the heterogeneous aspects of the reflector, for example, the vertical effects of materials in the horizontal access holes.

A brief analysis and description of the experiment reported previously⁽¹²⁾ served as a useful reference. A good deal of helpful information was obtained on a recent trip to the MTR site.

Three problems were treated in the analysis:

1. The statics of the reactor physics core.
2. The statics ($M_{wd} = 0$) of the power core.
3. The reactivity transient of the power core during burnup to 262 Mwd.

The reactor model for the statics problems was two-dimensional in the horizontal plane. For the burnup problem, an equivalent one-dimensional model was employed, the model approximating average conditions of core and reflector along a north-south line through the system.

Four-energy group diffusion theory was adopted for the reactor calculations. Group average cross sections and energy and spatial flux detail in the fuel elements was obtained in multi-group transport theory cell calculations.

Xenon and Samarium concentrations as a function of time were calculated assuming a uniform poison distribution in the core, and that power level changes are accomplished instantaneously.

Of the three major computations carried out in this study, two were completed with satisfactory results. The reactor model, basic data and computational methods appear to be quite adequate for reproducing the measured multiplication for both the reactor physics core and the ($M_{wd} = 0$) power core. The reactor physics system, based on calibrated shim rods, had a "rods-out" multiplication of 1.196. A computed value of 1.191 was obtained. The power core initial "rods-out" multiplication of 1.128 has been exactly duplicated. This is quite fortuitous since some further corrections of the order of 1.0% probably would be introduced in an even more detailed calculation. Plutonium cross sections based on a recent review by Sher⁽¹³⁾ improved this correlation by -3.0% $\Delta k/k$ over

(14)
computations using Leonard data .

The computation of the burnup reactivity transient for the power core remains unsatisfactory due to the inadequacies of the augmented one-dimensional model which was used (See Figure 9). Improvement in the calculation will largely depend upon how well the flux distortions resulting from partial shim rod insertions are approximated. Probably an explicit three-dimensional model is required.

FIGURE 9

MTR Cycle 108 Plutonium Burnup Experiment
Excess Reactivity versus Megawatt Days Operation

B. MTR Hx-Pu Fueling Proposal

Fairly extensive reactor statics and thermal-hydraulics investigations were carried out in order to scope a reasonable core design for the proposed MTR experiment. The details of the nuclear design calculations are contained in HW-84575⁽¹⁵⁾.

It was concluded that Pu-Al elements with reasonable compositions can probably be fabricated, while considerable R/D work would be required for alternate elements. It appears that up to 20 w/o Pu in Pu-Al can be fabricated. Burnup information (up to 90% burned) is available on 14 w/o U-Al elements.

Problems pertaining to reactor kinetics, control and hazards have not been investigated. Questions in these areas can generally be answered on the basis of information acquired during the 1958 MTR Lx-Pu experiment (Cycle 108). If a more comprehensive design evaluation is undertaken, these problems will have to be treated in greater detail.

Integration of the nuclear design, thermal-hydraulic, and materials considerations show that an MTR burnup experiment, utilizing Hx-Pu should be feasible.

To obtain substantial core lives with high Pu²⁴⁰ content fuels (~20 a/o Pu²⁴⁰), the present 9" wide MTR core should be augmented to 12" or 15". I.e., four or five rows of fuel elements should be employed. A 12" MTR core could probably be operated with about 13 kg of fuel (1. kg/lit. of core). The full power endurance (at 40 MW) for this core loading is estimated at about 3000 FPH. For the 15" core, the permissible core loading would be about 25 kg (.15/lit. of core) with an estimated endurance in the neighborhood of 6000 FPH. These endurance calculations were based on a Hx-Pu fuel containing 27 w/o Pu²⁴⁰.

Control of the highly loaded MTR cores could present problems. The use of burnable poisons may have to be considered. The use of soluble poisons as an auxiliary control device may also be of interest.

If physics experiments are carried on simultaneously, the presence of large beam holes in the reflector will result in reduction in initial k_{eff} and hence, reductions in endurance. For plutonium cores the thermal flux is more sharply peaked in the reflector than in comparable Uranium cores. However, the power density reduction for the larger Pu core, coupled with the higher Pu cross sections results in a net reduction of thermal flux levels in the reflector, compared to more lightly loaded ^{235}U cores.

V. Evaluation of APPR-type Reactor for a Possible Phoenix Fuel Power Experiment

In the latter part of FY 1965, some very preliminary work was started to evaluate the possibility of a full scale Phoenix power demonstration experiment. The APPR reactor served as a suitable model for these investigations. To date only nuclear analyses have been carried out. Material's problems and thermal-hydraulic analyses have as yet not been considered.

The APPR is a 10 MW_t, 140 liter pressurized water reactor fueled with $\text{U}^{235}\text{O}_2\text{-SS}$ cermet elements, clad with stainless steel. The L/D is 0.989 and the core M/W ratio is 0.262. The coolant water temperature is 450°F at 1200 psi. The core consists of a 7 x 7 box array with the corner boxes missing. The fuel boxes are 2.8" on a side and contain 18, 20 mil plates with a 5 mil clad. Seven of the boxes serve as control rods with 16 plate fuel box followers.

For purposes of the nuclear calculations an equivalent cylindrical model was postulated with an L/D of 0.989. An 18 cm radial water reflector was used, and a 5 cm axial end reflector savings was assumed. All calculations were performed

on this homogenized core, except for the flux depression calculations for the fuel plates. The thermal flux depression factors were obtained from a double- P_3 formalism. Burnup calculations were carried out by means of the ZODIAC burnup code.

To compare U-235 and Pu fueling, U-235 inventories from 10-40 Kg, and Pu inventories from 10-30 Kg were considered. Three types of Pu compositions were postulated, equivalent to those used in the PCTR experiments (See Section III-B) with Pu-240 contents of 8, 19, and 27 at%.

While this study is not yet complete some preliminary conclusions can be reached. If the low APPR M/W ratio of 0.262 is retained, a 19 a/o Pu-240 core loading of about 20 Kg might be of interest. This core would have an endurance of about 10000 FPH at 10 MW.

The large water fraction in the APPR adversely affects Pu-240 fertility and presumably modified designs with higher M/W's would be of more interest. The use of B-10 as a distributed burnable poison in the Pu core should also be considered. Such analysis will require more detailed nuclear as well as thermal-hydraulic calculations.

VI References

- (1) HW-79977 "Nuclear Characteristics of Some Compact, Water-Moderated Plutonium Burners" P. L. Hofmann, G. J. Busselman, R. H. Holeman, April 1964.
- (2) "Compact Water-Moderated Plutonium Burners" P. L. Hofmann, G. J. Busselman, R. H. Holeman. Trans. ANS Vol. 7 #1, June 1964.
- (3) BNWL-CC-119 "Comments on Pu Value in Hx-Pu Burners" P. L. Hofmann, J. B. Burnham. June 4, 1965.
- (4) P. F. Nichols, "Effective Resonance Integrals for Pu²⁴⁰ in Plutonium Aluminum Alloy Rods, " Nuclear Science and Engineering, Vol. 17, pages 144-148, (1963).
- (5) G. F. Kuncir, A Program for the Calculation of Resonance Integrals, GA-2525, August 28, 1961.
- (6) L. W. Nordheim, "The Theory of Resonance Absorption," Proceedings of Symposia in Applied Mathematics, Vol. XI (1959).
- (7) BNWL-97 "Theory-Experiment Comparison of Pu-240 Resonance Absorption" May 1965, C. M. Heeb.
- (8) HW-84369 Physics Quarterly Report, July-Sept, 1964. "UAL - Polyethylene Experiment in the PCTR" D. G. Albertson
- (9) BNWL-CC-148 "Physics Analysis of a UAL - Polyethylene System" G. J. Busselman. June 28, 1965.
- (10) HW-83431 "Analysis of MTR Cycle 108 - Low Exposure Plutonium Cores" J. J. Regimbal, November 19, 1964.
- (11) "Analysis of MTR Cycle 108 - Low Exposure Plutonium Cores" J. J. Regimbal, Trans. ANS Vol 7, #2, November 1964.
- (12) IDO-16508 "Operation of the MTR on a Plutonium Loading" D. G. de Boisblanc and R. S. Marsden, December 22, 1958.

- (13) BNL-722 "Least Squares Analysis for 2200 m/sec Parameters of U-233, U-235, and Pu-239" R. Sher, J. Felberbaum, June 1962.
- (14) "The Low Energy Cross Sections of Fissile Nuclides" B. R. Leonard, Jr.
Neutron Physics Proceedings of the Symposium held at Rensselaer Polytechnic Institute, May 5-6, 1961, New York Academic Press 1963, pg 3-34.
- (15) HW-84575 "Nuclear Design Calculations for a Hx-Pu Fueled MTR" R. H. Holeman, P. L. Hofmann, December 15, 1964.

ONSITE DISTRIBUTION

<u>Copy Number</u>	<u>Pacific Northwest Laboratory</u>
1	F. W. Albauth
2	D. G. Albertson
3	J. B. Burnham, Jr.
4	G. J. Busselman
5	F. G. Dawson
6	D. R. de Halas
7	E. A. Eschbach
8-18	J. C. Fox
19	R. W. Hardie
20	C. M. Heeb
21	R. E. Heineman
22-32	P. L. Hofmann
33	R. H. Holeman
34	D. D. Lanning
35	C. W. Lindenmeier
36	R. S. Paul
37	W. W. Porath
38	J. J. Regimbal
39	L. C. Schmid
40	K. G. Toyoda
41-45	Technical Information Files
46-47	Technical Publications

Richland Operations Office

48-49	L. R. Lucas
-------	-------------

OFFSITE DISTRIBUTION (SPECIAL)

2	General Atomic, San Diego, Calif. Attn: A. J. Goodjohn
6	Atomic Energy Commission, Washington Attn: M. J. Whitman, Chief Reactor Experiments Branch (2) I. F. Zartman, Chief Reactor Physics Branch (2) R. M. Grube, Water Reactors Branch (2)